UNITED STATES
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SUPPLEMENT 2 TO NUREG-0933

"A PRIORITIZATION OF GENERIC SAFETY ISSUES"

REVISION INSERTION INSTRUCTIONS

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TABLE II

LISTING OF ALL TMI ACTION PLAN ITEMS, TASK ACTION PLAN ITEMS, NEW GENERIC ISSUES, AND HUMAN FACTORS ISSUES

This table contains the priority designations for all issues listed in this report. For those issues found to be covered in other issues, the appropriate notations have been made in the Safety Priority Ranking column, e.g., I.A.2.2 in the Safety Priority Ranking column means that Item I.A.2.6(3) is covered in Item I.A.2.2. For resolved issues that have resulted in new requirements for operating plants, the appropriate multi-plant licensing action number is listed. The licensing action numbering system bears no relationship to the numbering systems used for identifying the prioritized issues. An explanation of the classification and status of the issues is provided in the legend below.

Legend

NOTES:
1 - Possible Resolution Identified for Evaluation
2 - Resolution Available (Documented in NUREG, NRC Memorandum, SER, or equivalent)
3 - Resolution Resulted in either: (a) The Establishment of New Regulatory Requirements (By Rule, SRP Change, or equivalent) or (b) No New Requirements
4 - Issue to be Prioritized in the Future

HIGH	- High Safety Priority
MEDIUM	- Medium Safety Priority
LOW	- Low Safety Priority
DROP	- Issue Dropped as a Generic Issue
E	- Environmental Issue
HFPP	- Human Factors Program Plan
I	- TMI Action Plan Item With Implementation of Resolution Mandated by NUREG-0737[98]
LI	- Licensing Issue
MPA	- Multi-Plant Action (See Status in NUREG-0748)[578]
NA	- Not Applicable
RI	- Regulatory Impact Issue
USI	- Unresolved Safety Issue (See Status in NUREG-0606)[60]

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
	TMI ACTION PLAN ITEMS						
I.A	OPERATING PERSONNEL						
I.A.1	Operating Personnel and Staffing						
I.A.1.1	Shift Technical Advisor	-	NRR/DHFS/LQB	I			F-01
I.A.1.2	Shift Supervisor Administrative Duties	-	NRR/DHFS/LQB	I			
I.A.1.3	Shift Manning	-	NRR/DHFS/LQB	I			F-02
I.A.1.4	Long-Term Upgrading	Colmar	RES/DFO/HFBR	NOTE 3(a)	1	6/30/84	
I.A.2	Training and Qualifications of Operating Personnel						
I.A.2.1	Immediate Upgrading of Operator and Senior Operator Training and Qualifications	-	-	-			
I.A.2.1(1)	Qualifications - Experience	-	NRR/DHFS/LQB	I			F-03
I.A.2.1(2)	Training	-	NRR/DHFS/LQB	I			F-03
I.A.2.1(3)	Facility Certification of Competence and Fitness of Applicants for Operator and Senior Operator Licenses	-	NRR/DHFS/LQB	I			F-03
I.A.2.2	Training and Qualifications of Operations Personnel	Colmar	NRR/DHFS/LQB	HFPP	1	12/31/84	NA
I.A.2.3	Administration of Training Programs	-	NRR/DHFS/LQB	I			
I.A.2.4	NRR Participation in Inspector Training	Colmar	NRR/DHFS/LQB	LI	1	12/31/84	NA
I.A.2.5	Plant Drills	Colmar	NRR/DHFS/LQB	LOW	1	12/31/84	NA
I.A.2.6	Long-Term Upgrading of Training and Qualifications	-	-	-			
I.A.2.6(1)	Revise Regulatory Guide 1.8	Colmar	NRR/DHFS/LQB	HFPP	1	12/31/84	NA
I.A.2.6(2)	Staff Review of NRR 80-117	Colmar	NRR/DHFS/LQB	NOTE 3(b)	1	12/31/84	NA
I.A.2.6(3)	Revise 10 CFR 55	Colmar	NRR/DHFS/LQB	I.A.2.2	1	12/31/84	NA
I.A.2.6(4)	Operator Workshops	Colmar	NRR/DHFS/LQB	MEDIUM	1	12/31/84	
I.A.2.6(5)	Develop Inspection Procedures for Training Program	Colmar	NRR/DHFS/LQB	NOTE 3(b)	1	12/31/84	NA
I.A.2.6(6)	Nuclear Power Fundamentals	Colmar	NRR/DHFS/LQB	DROP	1	12/31/84	NA
I.A.2.7	Accreditation of Training Institutions	Colmar	NRR/DHFS/LQB	HFPP	1	12/31/84	NA
I.A.3	Licensing and Requalification of Operating Personnel						
I.A.3.1	Revise Scope of Criteria for Licensing Examinations	Emrit	NRR/DHFS/LQB	I	2	12/31/84	
I.A.3.2	Operator Licensing Program Changes	Emrit	NRR/DHFS/OLB	NOTE 3(b)	2	12/31/84	NA
I.A.3.3	Requirements for Operator Fitness	Colmar	RES/DFO/HFBR	HFPP	2	12/31/84	NA
I.A.3.4	Licensing of Additional Operations Personnel	Thatcher	NRR/DHFS/LQB	HFPP	2	12/31/84	NA
I.A.3.5	Establish Statement of Understanding with INPO and DOE	Thatcher	NRR/DHFS/HFEB	LI (NOTE 3)	2	12/31/84	NA
I.A.4	Simulator Use and Development						
I.A.4.1	Initial Simulator Improvement	-	-	-			
I.A.4.1(1)	Short-Term Study of Training Simulators	Thatcher	NRR/DHFS/OLB	NOTE 3(b)	1	12/31/84	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
I.A.4.1(2)	Interim Changes in Training Simulators	Thatcher	NRR/DHFS/OLB	NOTE 3(a)	1	12/31/84	
I.A.4.2	Long-Term Training Simulator Upgrade	-	-	-			
I.A.4.2(1)	Research on Training Simulators	Colmar	RES/DHFS/OLB	HFPP	1	12/31/84	NA
I.A.4.2(2)	Upgrade Training Simulator Standards	Colmar	RES/DFO/HFBR	NOTE 3(a)	1	12/31/84	
I.A.4.2(3)	Regulatory Guide on Training Simulators	Colmar	RES/DFO/HFBR	NOTE 3(a)	1	12/31/84	
I.A.4.2(4)	Review Simulators for Conformance to Criteria	Colmar	NRR/DHFS/OLB	HFPP	1	12/31/84	NA
I.A.4.3	Feasibility Study of Procurement of NRC Training Simulator	Colmar	RES/DAE/RSRB	LI (NOTE 3)	1	12/31/84	NA
I.A.4.4	Feasibility Study of NRC Engineering Computer	Colmar	RES/DAE/RSRB	LI	1	12/31/84	NA
I.B.	SUPPORT PERSONNEL						
I.B.1	Management for Operations						
I.B.1.1	Organization and Management Long-Term Improvements	-	-	-			
I.B.1.1(1)	Prepare Draft Criteria	Colmar	NRR/DHFS/LQB	HFPP	1	12/31/84	NA
I.B.1.1(2)	Prepare Commission Paper	Colmar	NRR/DHFS/LQB	HFPP	1	12/31/84	NA
I.B.1.1(3)	Issue Requirements for the Upgrading of Management and Technical Resources	Colmar	NRR/DHFS/LQB	HFPP	1	12/31/84	NA
I.B.1.1(4)	Review Responses to Determine Acceptability	Colmar	NRR/DHFS/LQB	HFPP	1	12/31/84	NA
I.B.1.1(5)	Review Implementation of the Upgrading Activities	Colmar	OIE/DQASIP/ORPB	NOTE 3(b)	1	12/31/84	NA
I.B.1.1(6)	Prepare Revisions to Regulatory Guides 1.33 and 1.8	Colmar	NRR/DHFS/LQB	75, HFPP	1	12/31/84	NA
I.B.1.1(7)	Issue Regulatory Guides 1.33 and 1.8	Colmar	NRR/DHFS/LQB	75, HFPP	1	12/31/84	NA
I.B.1.2	Evaluation of Organization and Management Improvements of Near-Term Operating License Applicants	-	-	-			
I.B.1.2(1)	Prepare Draft Criteria	-	NRR/DHFS/LQB	I			
I.B.1.2(2)	Review Near-Term Operating License Facilities	-	NRR/DHFS/LQB	I			
I.B.1.2(3)	Include Findings in the SER for Each Near-Term Operating License Facility	-	NRR/DL/ORAB	I			
I.B.1.3	Loss of Safety Function	-	-	-			
I.B.1.3(1)	Require Licensees to Place Plant in Safest Shutdown Cooling Following a Loss of Safety Function Due to Personnel Error	Sege	RES	LI (NOTE 3)	1	12/31/84	NA
I.B.1.3(2)	Use Existing Enforcement Options to Accomplish Safest Shutdown Cooling	Sege	RES	LI (NOTE 3)	1	12/31/84	NA
I.B.1.3(3)	Use Non-Fiscal Approaches to Accomplish Safest Shutdown Cooling	Sege	RES	LI (NOTE 3)	1	12/31/84	NA
I.B.2	Inspection of Operating Reactors						
I.B.2.1	Revise OIE Inspection Program	-	-	-			
I.B.2.1(1)	Verify the Adequacy of Management and Procedural Controls and Staff Discipline	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.1(2)	Verify that Systems Required to Be Operable Are Properly Aligned	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
I.B.2.1(3)	Follow-up on Completed Maintenance Work Orders to Assure Proper Testing and Return to Service	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.1(4)	Observe Surveillance Tests to Determine Whether Test Instruments Are Properly Calibrated	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.1(5)	Verify that Licensees Are Complying with Technical Specifications	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.1(6)	Observe Routine Maintenance	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.1(7)	Inspect Terminal Boards, Panels, and Instrument Racks for Unauthorized Jumpers and Bypasses	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.2	Resident Inspector at Operating Reactors	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)		11/30/83	NA
I.B.2.3	Regional Evaluations	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)		11/30/83	NA
I.B.2.4	Overview of Licensee Performance	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)		11/30/83	NA
I.C	OPERATING PROCEDURES						
I.C.1	Short-Term Accident Analysis and Procedures Revision	-	-	-			
I.C.1(1)	Small Break LOCAs	-	NRR	I			
I.C.1(2)	Inadequate Core Cooling	-	NRR	I			
I.C.1(3)	Transients and Accidents	-	NRR	I			
I.C.1(4)	Confirmatory Analyses of Selected Transients	Riggs	NRR/DSI/RSB	NOTE 3(b)	1	12/31/84	NA
I.C.2	Shift and Relief Turnover Procedures	-	NRR	I			
I.C.3	Shift Supervisor Responsibilities	-	NRR	I			
I.C.4	Control Room Access	-	NRR	I			
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	-	NRR/DL	I			F-06
I.C.6	Procedures for Verification of Correct Performance of Operating Activities	-	NRR/DL	I			F-07
I.C.7	NSSS Vendor Review of Procedures	-	NRR/DHFS/PSRB	I			
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	-	NRR/DHFS/PSRB	I			
I.C.9	Long-Term Program Plan for Upgrading of Procedures	Riggs	NRR/DHFS/PSRB	HFPP	1	12/31/84	NA
I.D	CONTROL ROOM DESIGN						
I.D.1	Control Room Design Reviews	-	NRR/DL	I			F-08
I.D.2	Plant Safety Parameter Display Console	-	NRR/DL	I			F-09
I.D.3	Safety System Status Monitoring	Thatcher	NRR/DHFS/HFEB	HFPP	1	12/31/84	NA
I.D.4	Control Room Design Standard	Thatcher	NRR/DHFS/HFEB	HFPP	1	12/31/84	NA
I.D.5	Improved Control Room Instrumentation Research	-	-	-			
I.D.5(1)	Operator-Process Communication	Thatcher	RES/DFO/HFBR	NOTE 3(b)	1	12/31/84	NA
I.D.5(2)	Plant Status and Post-Accident Monitoring	Thatcher	RES/DFO/HFBR	NOTE 3(a)	1	12/31/84	
I.D.5(3)	On-Line Reactor Surveillance System	Thatcher	RES/DET/EEIGB	NOTE 1	1	12/31/84	

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
I.D.5(4)	Process Monitoring Instrumentation	Thatcher	RES/DFO/ICBR	NOTE 3(b)	1	12/31/84	NA
I.D.5(5)	Disturbance Analysis Systems	Thatcher	NRR/DHFS/HFEB	HFPP	1	12/31/84	NA
I.D.6	Technology Transfer Conference	Thatcher	RES/DFO/HFBR	LI (NOTE 3)	1	12/31/84	NA
I.E	ANALYSIS AND DISSEMINATION OF OPERATING EXPERIENCE						
I.E.1	Office for Analysis and Evaluation of Operational Data	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.2	Program Office Operational Data Evaluation	Matthews	NRR/DL/ORAB	LI (NOTE 3)	1	6/30/84	NA
I.E.3	Operational Safety Data Analysis	Matthews	RES/DRA/RRBR	LI (NOTE 3)	1	6/30/84	NA
I.E.4	Coordination of Licensee, Industry, and Regulatory Programs	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.5	Nuclear Plant Reliability Data System	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.6	Reporting Requirements	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.7	Foreign Sources	Matthews	IP	LI (NOTE 3)	1	6/30/84	NA
I.E.8	Human Error Rate Analysis	Matthews	RES/DFO/HFBR	LI (NOTE 3)	1	6/30/84	NA
I.F	QUALITY ASSURANCE						
I.F.1	Expand QA List	Pittman	OIE/DQASIP/QUAB	HIGH		11/30/83	
I.F.2	Develop More Detailed QA Criteria	-	-	-			
I.F.2(1)	Assure the Independence of the Organization Performing the Checking Function	Pittman	OIE/DQASIP/QUAB	LOW		11/30/83	NA
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)		11/30/83	NA
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)		11/30/83	NA
I.F.2(4)	Establish Criteria for Determining QA Requirements for Specific Classes of Equipment	Pittman	OIE/DQASIP/QUAB	LOW		11/30/83	NA
I.F.2(5)	Establish Qualification Requirements for QA and QC Personnel	Pittman	OIE/DQASIP/QUAB	LOW		11/30/83	NA
I.F.2(6)	Increase the Size of Licensees' QA Staff	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)		11/30/83	NA
I.F.2(7)	Clarify that the QA Program Is a Condition of the Construction Permit and Operating License	Pittman	OIE/DQASIP/QUAB	LOW		11/30/83	NA
I.F.2(8)	Compare NRC QA Requirements with Those of Other Agencies	Pittman	OIE/DQASIP/QUAB	LOW		11/30/83	NA
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)		11/30/83	NA
I.F.2(10)	Clarify Requirements for Maintenance of "As-Built" Documentation	Pittman	OIE/DQASIP/QUAB	LOW		11/30/83	NA
I.F.2(11)	Define Role of QA in Design and Analysis Activities	Pittman	OIE/DQASIP/QUAB	LOW		11/30/83	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
I.G	PREOPERATIONAL AND LOW-POWER TESTING						
I.G.1	Training Requirements	-	NRR/DHFS/PSRB	I			
I.G.2	Scope of Test Program	V'Molen	NRR/DHFS/PSRB	NOTE 3(a)	1	12/31/84	NA
II.A	SITING						
II.A.1	Siting Policy Reformulation	V'Molen	NRR/DE/SAB	NOTE 3(b)	1	12/31/84	NA
II.A.2	Site Evaluation of Existing Facilities	V'Molen	NRR/DE/SAB	V.A.1	1	12/31/84	NA
II.B	CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW						
II.B.1	Reactor Coolant System Vents	-	NRR/DL	I			F-10
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	-	NRR/DL	I			F-11
II.B.3	Post-Accident Sampling	-	NRR/DL	I			F-12
II.B.4	Training for Mitigating Core Damage	-	NRR/DL	I			F-13
II.B.5	Research on Phenomena Associated with Core Degradation and Fuel Melting	-	-	-			
II.B.5(1)	Behavior of Severely Damaged Fuel	V'Molen	RES/DAE/FBRB	HIGH		11/30/83	
II.B.5(2)	Behavior of Core Melt	V'Molen	RES/DAE/CSRB	HIGH		11/30/83	
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structure	V'Molen	RES/DAE/CSRB	MEDIUM		11/30/83	
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	Pittman	NRR/DST/RRAB	HIGH		11/30/83	
II.B.7	Analysis of Hydrogen Control	Matthews	NPR/DSI/CSB	II.B.8		11/30/83	
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	V'Molen	RES/ASTOP	HIGH		11/30/83	
II.C	RELIABILITY ENGINEERING AND RISK ASSESSMENT						
II.C.1	Interim Reliability Evaluation Program	Pittman	RES/DRA/RRBR	HIGH		11/30/83	
II.C.2	Continuation of Interim Reliability Evaluation Program	Pittman	NRR/DST/RRAB	HIGH		11/30/83	
II.C.3	Systems Interaction	Pittman	NRR/DST/GIB	A-17		11/30/83	
II.C.4	Reliability Engineering	Pittman	RES/DRA/RRBR	HIGH		11/30/83	
II.D	REACTOR COOLANT SYSTEM RELIEF AND SAFETY VALVES						
II.D.1	Testing Requirements	-	NRR/DL	I			F-14
II.D.2	Research on Relief and Safety Valve Test Requirements	Riggs	RES	LOW		11/30/83	NA
II.D.3	Relief and Safety Valve Position Indication	-	NRR	I			

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.E	SYSTEM DESIGN						
II.E.1	Auxiliary Feedwater System						
II.E.1.1	Auxiliary Feedwater System Evaluation	-	NRR/DL	I			F-15
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	-	NRR/DL	I			F-16, F-17
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	Riggs	RES/DRA/RRBR	NOTE 3(a)		11/30/83	
II.E.2	Emergency Core Cooling System						
II.E.2.1	Reliance on ECCS	Riggs	NRR/DSI/RSB	II.K.3(17)		11/30/83	NA
II.E.2.2	Research on Small Break LOCAs and Anomalous Transients	Riggs	RES/DAE/RSRB	MEDIUM		11/30/83	
II.E.2.3	Uncertainties in Performance Predictions	V'Molen	NRR/DSI/RSB	LOW		11/30/83	NA
II.E.3	Decay Heat Removal						
II.E.3.1	Reliability of Power Supplies for Natural Circulation	-	NRR	I			
II.E.3.2	Systems Reliability	V'Molen	NRR/DST/GIB	A-45		11/30/83	NA
II.E.3.3	Coordinated Study of Shutdown Heat Removal Requirements	V'Molen	NRR/DST/GIB	A-45		11/30/83	NA
II.E.3.4	Alternate Concepts Research	Riggs	RES/DAE/FBRB	NOTE 3(b)		11/30/83	NA
II.E.3.5	Regulatory Guide	Riggs	NRR/DST/GIB	A-45		11/30/83	NA
II.E.4	Containment Design						
II.E.4.1	Dedicated Penetrations	-	NRR/DL	I			F-18
II.E.4.2	Isolation Dependability	-	NRR/DL	I			F-19
II.E.4.3	Integrity Check	Milstead	NRR/DSI/CSB	HIGH		11/30/83	
II.E.4.4	Purging	-	-	-			
II.E.4.4(1)	Issue Letter to Licensees Requesting Limited Purging	Milstead	NRR/DSI/CSB	NOTE 3(a)		11/30/83	
II.E.4.4(2)	Issue Letter to Licensees Requesting Information on Isolation Letter	Milstead	NRR/DSI/CSB	NOTE 3(a)		11/30/83	
II.E.4.4(3)	Issue Letter to Licensees on Valve Operability	Milstead	NRR/DSI/CSB	NOTE 3(a)		11/30/83	
II.E.4.4(4)	Evaluate Purging and Venting During Normal Operation	Milstead	NRR/DSI/CSB	NOTE 3(b)		11/30/83	NA
II.E.4.4(5)	Issue Modified Purging and Venting Requirement	Milstead	NRR/DSI/CSB	NOTE 3(b)		11/30/83	NA
II.E.5	Design Sensitivity of B&W Reactors						
II.E.5.1	Design Evaluation	Thatcher	NRR/DSI/RSB	NOTE 3(a)	1	12/31/84	
II.E.5.2	B&W Reactor Transient Response Task Force	Thatcher	NRR/DL/ORAB	NOTE 3(a)	1	12/31/84	
II.E.6	In Situ Testing of Valves						
II.E.6.1	Test Adequacy Study	Thatcher	NRR/DE/EQB	MEDIUM		11/30/83	

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.F	INSTRUMENTATION AND CONTROLS						
II.F.1	Additional Accident Monitoring Instrumentation	-	NRR/DL	I			F-20, F-21, F-22, F-23, F-24, F-25
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	-	NRR/DL	I			F-26
II.F.3	Instruments for Monitoring Accident Conditions	V'Molen	RES/DFO/ICBR	NOTE 3(a)		11/30/83	
II.F.4	Study of Control and Protective Action Design Requirements	Thatcher	NRR/DSI/ICSB	DROP		11/30/83	NA
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	Thatcher	RES/DFO/ICBR	MEDIUM		11/30/83	
II.G	ELECTRICAL POWER						
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	-	NRR	I			
II.H	TMI-2 CLEANUP AND EXAMINATION						
II.H.1	Maintain Safety of TMI-2 and Minimize Environmental Impact	Matthews	NRR/TMIPO	NOTE 3(b)		11/30/83	NA
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	Milstead	RES/DAE/FBRB	HIGH		11/30/83	
II.H.3	Evaluate and Feed Back Information Obtained from TMI	Milstead	NRR/TMIPO	II.H.2		11/30/83	NA
II.H.4	Determine Impact of TMI on Socioeconomic and Real Property Values	Milstead	RES/DHSWM/SEBR	LI (NOTE 3)		11/30/83	NA
II.J	GENERAL IMPLICATIONS OF TMI FOR DESIGN AND CONSTRUCTION ACTIVITIES						
II.J.1	Vendor Inspection Program						
II.J.1.1	Establish a Priority System for Conducting Vendor Inspections	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.1.2	Modify Existing Vendor Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.1.4	Assign Resident Inspectors to Reactor Vendors and Architect-Engineers	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.J.2	Construction Inspection Program						
II.J.2.1	Reorient Construction Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.2.2	Increase Emphasis on Independent Measurement in Construction Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.2.3	Assign Resident Inspectors to All Construction Sites	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.3	Management for Design and Construction						
II.J.3.1	Organization and Staffing to Oversee Design and Construction	Pittman	NRR/DHFS/LQB	I.B.1.1		11/30/83	NA
II.J.3.2	Issue Regulatory Guide	Pittman	NRR/DHFS/LQB	I.B.1.1		11/30/83	NA
II.J.4	Revise Deficiency Reporting Requirements						
II.J.4.1	Revise Deficiency Reporting Requirements	Riani	RES/DRA/RABR	NOTE 2		11/30/83	
II.K	MEASURES TO MITIGATE SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS AND LOSS-OF-FEEDWATER ACCIDENTS						
II.K.1	IE Bulletins	-	-	-			
II.K.1(1)	Review TMI-2 PNs and Detailed Chronology of the TMI-2 Accident	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(2)	Review Transients Similar to TMI-2 That Have Occurred at Other Facilities and NRC Evaluation of Davis-Besse Event	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(4)	Review Operating Procedures and Training Instructions	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(5)	Safety-Related Valve Position Description	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(7)	Implement Positive Position Controls on Valves That Could Compromise or Defeat AFW Flow	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(9)	Review Procedures to Assure That Radioactive Liquids and Gases Are Not Transferred out of Containment Inadvertently	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(11)	Make All Operating and Maintenance Personnel Aware of the Seriousness and Consequences of the Erroneous Actions Leading up to, and in Early Phases of, the TMI-2 Accident	Emrit	NRR	NOTE 3(a)		12/31/84	-

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.K.1(12)	One Hour Notification Requirement and Continuous Communications Channels	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(14)	Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(15)	For Facilities with Non-Automatic AFW Initiation, Provide Dedicated Operator in Continuous Communication with CR to Operate AFW	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(16)	Implement Procedures That Identify PRZ PORV "Open" Indications and That Direct Operator to Close Manually at "Reset" Setpoint	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(17)	Trip PZR Level Bistable so That PZR Low Pressure Will Initiate Safety Injection	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(18)	Develop Procedures and Train Operators on Methods of Establishing and Maintaining Natural Circulation	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(19)	Describe Design and Procedure Modifications to Reduce Likelihood of Automatic PZR PORV Actuation in Transients	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(20)	Provide Procedures and Training to Operators for Prompt Manual Reactor Trip for LOFW, TT, MSIV Closure, LOOP, LOSG Level, and LO PZR Level	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(21)	Provide Automatic Safety-Grade Anticipatory Reactor Trip for LOFW, TT, or Significant Decrease in SG Level	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(22)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(23)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(24)	Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses Between Reactor Trip and RCP Trip	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(25)	Develop Operator Action Guidelines	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(26)	Revise Emergency Procedures and Train ROs and SROs	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(27)	Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.2	Commission Orders on B&W Plants	-	-	-			
II.K.2(1)	Upgrade Timeliness and Reliability of AFW System	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(2)	Procedures and Training to Initiate and Control AFW Independent of Integrated Control System	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.2(3)	Hard-Wired Control-Grade Anticipatory Reactor Trips	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(4)	Small-Break LOCA Analysis, Procedures and Operator Training	Emrit	NRR/DHFS/OLB	NOTE 3(a)		12/31/84	-

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.2(6)	Reevaluate Analysis for Dual-Level Setpoint Control	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(7)	Reevaluate Transient of September 24, 1977	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(8)	Continued Upgrading of AFW System	Emrit	NRR	II.E.1.1, II.E.1.2		12/31/84	NA
II.K.2(9)	Analysis and Upgrading of Integrated Control System	Emrit	NRR	I		12/31/84	F-27
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	Emrit	NRR	I		12/31/84	F-28
II.K.2(11)	Operator Training and Drilling	Emrit	NRR	I		12/31/84	F-29
II.K.2(12)	Transient Analysis and Procedures for Management of Small Breaks	Emrit	NRR	I.C.1		12/31/84	NA
II.K.2(13)	Thermal-Mechanical Report on Effect of HPI on Vessel Integrity for Small-Break LOCA With No AFW	Emrit	NRR	I		12/31/84	F-30
II.K.2(14)	Demonstrate That Predicted Lift Frequency of PORVs and SVs Is Acceptable	Emrit	NRR	I		12/31/84	F-31
II.K.2(15)	Analysis of Effects of Slug Flow on Once-Through Steam Generator Tubes After Primary System Voiding	Emrit	NRR	I		12/31/84	-
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA With Loss of Offsite Power	Emrit	NRR	I		12/31/84	F-32
II.K.2(17)	Analysis of Potential Voiding in RCS During Anticipated Transients	Emrit	NRR	I		12/31/84	F-33
II.K.2(18)	Analysis of Loss of Feedwater and Other Anticipated Transients	Emrit	NRR	I.C.1		12/31/84	NA
II.K.2(19)	Benchmark Analysis of Sequential AFW Flow to Once-Through Steam Generator	Emrit	NRR	I		12/31/84	F-34
II.K.2(20)	Analysis of Steam Response to Small-Break LOCA That Causes System Pressure to Exceed PORV Setpoint	Emrit	NRR	I		12/31/84	F-35
II.K.2(21)	LOFT L3-1 Predictions	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.3	Final Recommendations of Bulletins and Orders Task Force	-	-	-			
II.K.3(1)	Install Automatic PORV Isolation System and Perform Operational Test	Emrit	NRR	I		12/31/84	F-36
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	Emrit	NRR	I		12/31/84	F-37
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	Emrit	NRR	I		12/31/84	F-38
II.K.3(4)	Review and Upgrade Reliability and Redundancy of Non-Safety Equipment for Small-Break LOCA Mitigation	Emrit	NRR	II.C.1, II.C.2, II.C.3		12/31/84	NA
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps	Emrit	NRR	I		12/31/84	F-39
II.K.3(6)	Instrumentation to Verify Natural Circulation	Emrit	NRR/DSI	I.C.1, II.F.2, II.F.3		12/31/84	NA
II.K.3(7)	Evaluation of PORV Opening Probability During Overpressure Transient	Emrit	NRR	I		12/31/84	-

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.K.3(8)	Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of SGs	Emrit	NRR/DST/GIB	II.C.1, II.E.3.3		12/31/84	NA
II.K.3(9)	Proportional Integral Derivative Controller Modification	Emrit	NRR	I		12/31/84	F-40
II.K.3(10)	Anticipatory Trip Modification Proposed by Some Licensees to Confine Range of Use to High Power Levels	Emrit	NRR	I		12/31/84	F-41
II.K.3(11)	Control Use of PORV Supplied by Control Components, Inc. Until Further Review Complete	Emrit	NRR	I		12/31/84	-
II.K.3(12)	Confirm Existence of Anticipatory Trip Upon Turbine Trip	Emrit	NRR	I		12/31/84	F-42
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	Emrit	NRR	I		12/31/84	F-43
II.K.3(14)	Isolation of Isolation Condensers on High Radiation	Emrit	NRR	I		12/31/84	F-44
II.K.3(15)	Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	Emrit	NRR	I		12/31/84	F-45
II.K.3(16)	Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification	Emrit	NRR	I		12/31/84	F-46
II.K.3(17)	Report on Outage of ECC Systems - Licensee Report and Technical Specification Changes	Emrit	NRR	I		12/31/84	F-47
II.K.3(18)	Modification of ADS Logic - Feasibility Study and Modification for Increased Diversity for Some Event Sequences	Emrit	NRR	I		12/31/84	F-48
II.K.3(19)	Interlock on Recirculation Pump Loops	Emrit	NRR	I		12/31/84	F-49
II.K.3(20)	Loss of Service Water for Big Rock Point	Emrit	NRR	I		12/31/84	-
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level - Design and Modification	Emrit	NRR	I		12/31/84	F-50
II.K.3(22)	Automatic Switchover of RCIC System Suction - Verify Procedures and Modify Design	Emrit	NRR	I		12/31/84	F-51
II.K.3(23)	Central Water Level Recording	Emrit	NRR	I.D.2, III.A.1.2, III.A.3.4		12/31/84	NA
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	Emrit	NRR	I		12/31/84	F-52
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	Emrit	NRR	I		12/31/84	F-53
II.K.3(26)	Study Effect on RHR Reliability of Its Use for Fuel Pool Cooling	Emrit	NRR/DSI	II.E.2.1		12/31/84	NA
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	Emrit	NRR	I		12/31/84	F-54
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	Emrit	NRR	I		12/31/84	F-55
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles	Emrit	NRR	I		12/31/84	F-56
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	Emrit	NRR	I		12/31/84	F-57
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	Emrit	NRR	I		12/31/84	F-58

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.K.3(32)	Provide Experimental Verification of Two-Phase Natural Circulation Models	Emrit	NRR/DSI	II.E.2.2		12/31/84	NA
II.K.3(33)	Evaluate Elimination of PORV Function	Emrit	NRR	II.C.1		12/31/84	NA
II.K.3(34)	Relap-4 Model Development	Emrit	NRR/DSI	II.E.2.2		12/31/84	NA
II.K.3(35)	Evaluation of Effects of Core Flood Tank Injection on Small-Break LOCAs	Emrit	NRR	I.C.1		12/31/84	NA
II.K.3(36)	Additional Staff Audit Calculations of B&W Small-Break LOCA Analyses	Emrit	NRR	I.C.1		12/31/84	NA
II.K.3(37)	Analysis of B&W Response to Isolated Small-Break LOCA	Emrit	NRR	I.C.1		12/31/84	NA
II.K.3(38)	Analysis of Plant Response to a Small-Break LOCA in the Pressurizer Spray Line	Emrit	NRR	I.C.1		12/31/84	NA
II.K.3(39)	Evaluation of Effects of Water Slugs in Piping Caused by HPI and CFT Flows	Emrit	NRR	I.C.1		12/31/84	NA
II.K.3(40)	Evaluation of RCP Seal Damage and Leakage During a Small-Break LOCA	Emrit	NRR	II.K.2(16)		12/31/84	NA
II.K.3(41)	Submit Predictions for LOFT Test L3-6 with RCPs Running	Emrit	NRR	I.C.1		12/31/84	NA
II.K.3(42)	Submit Requested Information on the Effects of Non-Condensible Gases	Emrit	NRR	I.C.1		12/31/84	NA
II.K.3(43)	Evaluation of Mechanical Effects of Slug Flow on Steam Generator Tubes	Emrit	NRR	II.K.2(15)		12/31/84	NA
II.K.3(44)	Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	Emrit	NRR	I		12/31/84	F-59
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	Emrit	NRR	I		12/31/84	F-60
II.K.3(46)	Response to List of Concerns from ACRS Consultant	Emrit	NRR	I		12/31/84	F-61
II.K.3(47)	Test Program for Small-Break LOCA Model Verification Pretest Prediction, Test Program, and Model Verification	Emrit	NRR	I.C.1, II.E.2.2		12/31/84	NA
II.K.3(48)	Assess Change in Safety Reliability as a Result of Implementing B&OTF Recommendations	Emrit	NRR	II.C.1, II.C.2		12/31/84	NA
II.K.3(49)	Review of Procedures (NRC)	Emrit	NRR/DHFS/PSRB	I.C.8, I.C.9		12/31/84	NA
II.K.3(50)	Review of Procedures (NSSS Vendors)	Emrit	NRR/DHFS/PSRB	I.C.7, I.C.9		12/31/84	NA
II.K.3(51)	Symptom-Based Emergency Procedures	Emrit	NRR/DHFS/PSRB	I.C.9		12/31/84	NA
II.K.3(52)	Operator Awareness of Revised Emergency Procedures	Emrit	NRR	I.B.1.1, I.C.2, I.C.5		12/31/84	NA
II.K.3(53)	Two Operators in Control Room	Emrit	NRR	I.A.1.3		12/31/84	NA
II.K.3(54)	Simulator Upgrade for Small-Break LOCAs	Emrit	NRR	I.A.4.1		12/31/84	NA
II.K.3(55)	Operator Monitoring of Control Board	Emrit	NRR	I.C.1, I.D.2, I.D.3		12/31/84	NA
II.K.3(56)	Simulator Training Requirements	Emrit	NRR/DHFS/OLB	I.A.2.6, I.A.3.1		12/31/84	NA
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS	Emrit	NRR	I		12/31/84	F-62

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Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
III.A	EMERGENCY PREPAREDNESS AND RADIATION EFFECTS						
III.A.1	Improve Licensee Emergency Preparedness - Short Term						
III.A.1.1	Upgrade Emergency Preparedness	-	-	-			
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	-	OIE/DEPER/EPB	I			
III.A.1.1(2)	Perform an Integrated Assessment of the Implementation	-	OIE/DEPER/EPB	I			
III.A.1.2	Upgrade Licensee Emergency Support Facilities	-	-	-			
III.A.1.2(1)	Technical Support Center	-	OIE/DEPER/EPB	I			F-63
III.A.1.2(2)	On-Site Operational Support Center	-	OIE/DEPER/EPB	I			F-64
III.A.1.2(3)	Near-Site Emergency Operations Facility	-	OIE/DEPER/EPB	I			F-65
III.A.1.3	Maintain Supplies of Thyroid-Blocking Agent	-	-	-			
III.A.1.3(1)	Workers	Riggs	OIE/DEPER/EPB	NOTE 3(b)		11/30/83	NA
III.A.1.3(2)	Public	Riggs	OIE/DEPER/EPB	NOTE 1		11/30/83	
III.A.2	Improving Licensee Emergency Preparedness-Long Term						
III.A.2.1	Amend 10 CFR 50 and 10 CFR 50, Appendix E	-	-	-			
III.A.2.1(1)	Publish Proposed Amendments to the Rules	-	RES	I			
III.A.2.1(2)	Conduct Public Regional Meetings	-	RES	I			
III.A.2.1(3)	Prepare Final Commission Paper Recommending Adoption of Rules	-	RES	I			
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	-	OIE	I			F-67
III.A.2.2	Development of Guidance and Criteria	-	NRR/DL	I			F-68
III.A.3	Improving NRC Emergency Preparedness						
III.A.3.1	NRC Role in Responding to Nuclear Emergencies	-	-	-			
III.A.3.1(1)	Define NRC Role in Emergency Situations	Riggs	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.A.3.1(2)	Revise and Upgrade Plans and Procedures for the NRC Emergency Operations Center	Riggs	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.A.3.1(3)	Revise Manual Chapter 0502, Other Agency Procedures, and NUREG-0610	Riggs	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.A.3.1(4)	Prepare Commission Paper	Riggs	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.A.3.1(5)	Revise Implementing Procedures and Instructions for Regional Offices	Riggs	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.A.3.2	Improve Operations Centers	Riggs	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.A.3.3	Communications	-	-	-			
III.A.3.3(1)	Install Direct Dedicated Telephone Lines	Pittman	OIE/DEPER/IRDB	NOTE 3(a)		11/30/83	NA
III.A.3.3(2)	Obtain Dedicated, Short-Range Radio Communication Systems	Pittman	OIE/DEPER/IRDB	NOTE 3(a)		11/30/83	NA
III.A.3.4	Nuclear Data Link	Thatcher	OIE/DEPER/IRDB	MEDIUM		11/30/83	
III.A.3.5	Training, Drills, and Tests	Pittman	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.A.3.6	Interaction of NRC and Other Agencies	-	-	-			
III.A.3.6(1)	International	Pittman	OIE/DEPER/EPLB	NOTE 3(b)		11/30/83	NA
III.A.3.6(2)	Federal	Pittman	OIE/DEPER/EPLB	NOTE 3(b)		11/30/83	NA
III.A.3.6(3)	State and Local	Pittman	OIE/DEPER/EPLB	NOTE 3(b)		11/30/83	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
III.B	EMERGENCY PREPAREDNESS OF STATE AND LOCAL GOVERNMENTS						
III.B.1	Transfer of Responsibilities to FEMA	Milstead	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.B.2	Implementation of NRC and FEMA Responsibilities	-	-	-			
III.B.2(1)	The Licensing Process	Milstead	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.B.2(2)	Federal Guidance	Milstead	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.C	PUBLIC INFORMATION						
III.C.1	Have Information Available for the News Media and the Public	-	-	-			
III.C.1(1)	Review Publicly Available Documents	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.1(2)	Recommend Publication of Additional Information	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.1(3)	Program of Seminars for News Media Personnel	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.2	Develop Policy and Provide Training for Interfacing With the News Media	-	-	-			
III.C.2(1)	Develop Policy and Procedures for Dealing With Briefing Requests	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.2(2)	Provide Training for Members of the Technical Staff	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.D	RADIATION PROTECTION						
III.D.1	Radiation Source Control						
III.D.1.1	Primary Coolant Sources Outside the Containment Structure	-	-	-			
III.D.1.1(1)	Review Information Submitted by Licensees Pertaining to Reducing Leakage from Operating Systems	-	NRR	I			
III.D.1.1(2)	Review Information on Provisions for Leak Detection	Emrit	NRR/DSI/METB	NOTE 4			
III.D.1.1(3)	Develop Proposed System Acceptance Criteria	Emrit	NRR/DSI/METB	NOTE 4			
III.D.1.2	Radioactive Gas Management	Emrit	NRR/DSI/METB	DROP		11/30/83	NA
III.D.1.3	Ventilation System and Radioiodine Adsorber Criteria	-	-	-			
III.D.1.3(1)	Decide Whether Licensees Should Perform Studies and Make Modifications	Emrit	NRR/DSI/METB	DROP		11/30/83	NA
III.D.1.3(2)	Review and Revise SRP	Emrit	NRR/DSI/METB	DROP		11/30/83	NA
III.D.1.3(3)	Require Licensees to Upgrade Filtration Systems	Emrit	NRR/DSI/METB	DROP		11/30/83	NA
III.D.1.3(4)	Sponsor Studies to Evaluate Charcoal Adsorber	Emrit	NRR/DSI/METB	NOTE 3(b)		11/30/83	NA
III.D.1.4	Radwaste System Design Features to Aid in Accident Recovery and Decontamination	Emrit	NRR/DSI/METB	DROP		11/30/83	NA
III.D.2	Public Radiation Protection Improvement						
III.D.2.1	Radiological Monitoring of Effluents	-	-	-			
III.D.2.1(1)	Evaluate the Feasibility and Perform a Value-Impact Analysis of Modifying Effluent-Monitoring Design Criteria	Emrit	NRR/DSI/METB	LOW	1	6/30/84	NA

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Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
III.D.2.1(2)	Study the Feasibility of Requiring the Development of Effective Means for Monitoring and Sampling Noble Gases and Radioiodine Released to the Atmosphere	Emrit	NRR/DSI/METB	LOW	1	6/30/84	NA
III.D.2.1(3)	Revise Regulatory Guides	Emrit	NRR/DSI/METB	LOW	1	6/30/84	NA
III.D.2.2	Radioiodine, Carbon-14, and Tritium Pathway Dose Analysis	-	-	-			
III.D.2.2(1)	Perform Study of Radioiodine, Carbon-14, and Tritium Behavior	Emrit	NRR/DSI/RAB	NOTE 3(b)	1	6/30/84	NA
III.D.2.2(2)	Evaluate Data Collected at Quad Cities	Emrit	NRR/DSI/RAB	III.D.2.5	1	6/30/84	NA
III.D.2.2(3)	Determine the Distribution of the Chemical Species of Radioiodine in Air-Water-Steam Mixtures	Emrit	NRR/DSI/RAB	III.D.2.5	1	6/30/84	NA
III.D.2.2(4)	Revise SRP and Regulatory Guides	Emrit	NRR/DSI/RAB	III.D.2.5	1	6/30/84	NA
III.D.2.3	Liquid Pathway Radiological Control	-	-	-			
III.D.2.3(1)	Develop Procedures to Discriminate Between Sites/Plants	Emrit	NRR/DE/EHEB	NOTE 1	1	6/30/84	
III.D.2.3(2)	Discriminate Between Sites and Plants That Require Consideration of Liquid Pathway Interdiction Techniques	Emrit	NRR/DE/EHEB	NOTE 1	1	6/30/84	
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction	Emrit	NRR/DE/EHEB	NOTE 1	1	6/30/84	
III.D.2.3(4)	Prepare a Summary Assessment	Emrit	NRR/DE/EHEB	NOTE 1	1	6/30/84	
III.D.2.4	Offsite Dose Measurements	-	-	-			
III.D.2.4(1)	Study Feasibility of Environmental Monitors	V'Molen	NRR/DSI/RAB	NOTE 3(b)	1	6/30/84	NA
III.D.2.4(2)	Place 50 TLDs Around Each Site	V'Molen	OIE/DRP/ORPB	LI (NOTE 3)	1	6/30/84	NA
III.D.2.5	Offsite Dose Calculation Manual	V'Molen	NRR/DSI/RAB	NOTE 3(b)	1	6/30/84	
III.D.2.6	Independent Radiological Measurements	V'Molen	OIE/DRP/ORPB	LI (NOTE 3)	1	6/30/84	NA
<u>III.D.3</u>	<u>Worker Radiation Protection Improvement</u>						
III.D.3.1	Radiation Protection Plans	V'Molen	NRR/DSI/RAB	HIGH		11/30/83	
III.D.3.2	Health Physics Improvements	-	-	-			
III.D.3.2(1)	Amend 10 CFR 20	V'Molen	RES/DFO/ORPBR	LI (NOTE 2)		11/30/83	NA
III.D.3.2(2)	Issue a Regulatory Guide	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)		11/30/83	NA
III.D.3.2(3)	Develop Standard Performance Criteria	V'Molen	RES/DFO/ORPBR	LI (NOTE 2)		11/30/83	NA
III.D.3.2(4)	Develop Method for Testing and Certifying Air-Purifying Respirators	V'Molen	RES/DFO/ORPBR	LI (NOTE 2)		11/30/83	NA
III.D.3.3	Inplant Radiation Monitoring	-	-	-			
III.D.3.3(1)	Issue Letter Requiring Improved Radiation Sampling Instrumentation	-	NRR/DL	I			F-69
III.D.3.3(2)	Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment	-	NRR	I			
III.D.3.3(3)	Issue a Rule Change Providing Acceptable Methods for Calibration of Radiation-Monitoring Instruments	-	RES	I			
III.D.3.3(4)	Issue a Regulatory Guide	-	RES	I			
III.D.3.4	Control Room Habitability	-	NRR/DL	I			F-70
III.D.3.5	Radiation Worker Exposure	-	-	-			
III.D.3.5(1)	Develop Format for Data To Be Collected by Utilities Regarding Total Radiation Exposure to Workers	V'Molen	RES/DFO/ORPBR	LI		11/30/83	NA
III.D.3.5(2)	Investigative Methods of Obtaining Employee Health Data by Nonlegislative Means	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)		11/30/83	NA
III.D.3.5(3)	Revise 10 CFR 20	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)		11/30/83	NA

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Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
IV.A	STRENGTHEN ENFORCEMENT PROCESS						
IV.A.1	Seek Legislative Authority	Emrit	GC	LI (NOTE 3)		11/30/83	NA
IV.A.2	Revise Enforcement Policy	Emrit	OIE/ES	LI (NOTE 3)		11/30/83	NA
IV.B	ISSUANCE OF INSTRUCTIONS AND INFORMATION TO LICENSEES						
IV.B.1	Revise Practices for Issuance of Instructions and Information to Licensees	Emrit	OIE/DEPER	LI (NOTE 3)		11/30/83	NA
IV.C	EXTEND LESSONS LEARNED TO LICENSED ACTIVITIES OTHER THAN POWER REACTORS						
IV.C.1	Extend Lessons Learned from TMI to Other NRC Programs	Emrit	NMSS/WM	NOTE 3(b)		11/30/83	NA
IV.D	NRC STAFF TRAINING						
IV.D.1	NRC Staff Training	Emrit	ADM/MDTS	LI (NOTE 3)		11/30/83	NA
IV.E	SAFETY DECISION-MAKING						
IV.E.1	Expand Research on Quantification of Safety Decision-Making	Colmar	RES/DRA/RABR	LI		11/30/83	NA
IV.E.2	Plan for Early Resolution of Safety Issues	Emrit	NRR/DST/SPEB	LI (NOTE 3)		11/30/83	NA
IV.E.3	Plan for Resolving Issues at the CP Stage	Colmar	RES/DRA/RABR	LI (NOTE 2)		11/30/83	NA
IV.E.4	Resolve Generic Issues by Rulemaking	Colmar	RES/DRA/RABR	LI		11/30/83	NA
IV.E.5	Assess Currently Operating Reactors	Matthews	NRR/DL/SEPB	HIGH		11/30/83	
IV.F	FINANCIAL DISINCENTIVES TO SAFETY						
IV.F.1	Increased OIE Scrutiny of the Power-Ascension Test Program	Thatcher	OIE/DQASIP	NOTE 3(b)		11/30/83	NA
IV.F.2	Evaluate the Impacts of Financial Disincentives to the Safety of Nuclear Power Plants	Matthews	SP	NOTE 3(b)		11/30/83	NA

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IV.G	IMPROVE SAFETY RULEMAKING PROCEDURES						
IV.G.1	Develop a Public Agenda for Rulemaking	Emrit	ADM/RPB	LI (NOTE 3)		11/30/83	NA
IV.G.2	Periodic and Systematic Reevaluation of Existing Rules	Milstead	RES/DRA/RABR	LI		11/30/83	NA
IV.G.3	Improve Rulemaking Procedures	Milstead	RES/DRA/RABR	LI (NOTE 3)		11/30/83	NA
IV.G.4	Study Alternatives for Improved Rulemaking Process	Milstead	RES/DRA/RABR	LI (NOTE 3)		11/30/83	NA
IV.H	NRC PARTICIPATION IN THE RADIATION POLICY COUNCIL						
IV.H.1	NRC Participation in the Radiation Policy Council	Sege	RES/DHSWM/HEBR	LI (NOTE 3)		11/30/83	NA
	TASK ACTION PLAN ITEMS						
A-1	Water Hammer	-	NRR/DST/GIB	USI		11/30/83	
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	-	NRR/DST/GIB	USI		11/30/83	D-10
A-3	Westinghouse Steam Generator Tube Integrity	-	NRR/DST/GIB	USI		11/30/83	
A-4	CE Steam Generator Tube Integrity	-	NRR/DST/GIB	USI		11/30/83	
A-5	B&W Steam Generator Tube Integrity	-	NRR/DST/GIB	USI		11/30/83	
A-6	Mark I Short Term Program	-	NRR/DST/GIB	USI		11/30/83	
A-7	Mark I Long Term Program	-	NRR/DST/GIB	USI		11/30/83	D-01
A-8	Mark II Containment Pool Dyanmic Loads - Long Term Program	-	NRR/DST/GIB	USI		11/30/83	
A-9	ATWS	-	NRR/DST/GIB	USI		11/30/83	
A-10	BWR Feedwater Nozzle Cracking	-	NRR/DST/GIB	USI		11/30/83	B-25
A-11	Reactor Vessel Materials Toughness	-	NRR/DST/GIB	USI		11/30/83	
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	-	NRR/DST/GIB	USI		11/30/83	
A-13	Snubber Operability Assurance	Emrit	NRR/DE/MEB	NOTE 3(a)		11/30/83	
A-14	Flaw Detection	Matthews	NRR/DE/MTEB	DROP		11/30/83	NA
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning	Pittman	NRR/DE/CHEB	NOTE 3(b)		11/30/83	NA
A-16	Steam Effects on BWR Core Spray Distribution	Emrit	NRR/DSI/CPB	NOTE 3(a)		11/30/83	D-12
A-17	Systems Interaction	-	NRR/DST/GIB	USI		11/30/83	
A-18	Pipe Rupture Design Criteria	Emrit	NRR/DE/MEB	DROP		11/30/83	NA
A-19	Digital Computer Protection System	Thatcher	NRR/DSI/ICSB	NOTE 4		11/30/83	
A-20	Impacts of the Coal Fuel Cycle	-	NRR/DE/EHEB	LI		11/30/83	NA
A-21	Main Steamline Break Inside Containment - Evaluation of Environmental Conditions for Equipment Qualification	V'Molen	NRR/DSI/CSB	LOW		11/30/83	NA

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Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
A-22	PWR Main Steamline Break - Core, Reactor Vessel and Containment Building Response	V'Molen	NRR/DSI/CSB	DROP		11/30/83	NA
A-23	Containment Leak Testing	Matthews	NRR/DSI/CSB	RI		11/30/83	
A-24	Qualification of Class 1E Safety Related Equipment	-	NRR/DST/GIB	USI		11/30/83	B-60
A-25	Non-Safety Loads on Class 1E Power Sources	Thatcher	NRR/DSI/PSB	NOTE 3(a)		11/30/83	
A-26	Reactor Vessel Pressure Transient Protection	-	NRR/DST/GIB	USI		11/30/83	B-04
A-27	Reload Applications	-	NRR/DSI/CPB	LI		11/30/83	NA
A-28	Increase in Spent Fuel Pool Storage Capacity	Colmar	NRR/DE/SGEB	NOTE 3(a)		11/30/83	
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	Colmar	NRR/DSI/ASB	MEDIUM		11/30/83	
A-30	Adequacy of Safety-Related DC Power Supplies	Sege	NRR/DSI/PSB	HIGH		11/30/83	
A-31	RHR Shutdown Requirements	-	NRR/DST/GIB	USI		11/30/83	
A-32	Missile Effects	Pittman	NRR/DE/MTEB	A-37, A-38, B-68		11/30/83	NA
A-33	NEPA Review of Accident Risks	-	NRR/DSI/AEB	E(NOTE 3)		11/30/83	NA
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents	V'Molen	NRR/DSI/ICSB	II.F.3		11/30/83	NA
A-35	Adequacy of Offsite Power Systems	Emrit	NRR/DSI/PSB	NOTE 3(a)		11/30/83	
A-36	Control of Heavy Loads Near Spent Fuel	-	NRR/DSI/GIB	USI		11/30/83	C-10, C-15
A-37	Turbine Missiles	Pittman	NRR/DE/MTEB	DROP		11/30/83	NA
A-38	Tornado Missiles	Sege	NRR/DSI/ASB	LOW		11/30/83	NA
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits	-	NRR/DST/GIB	USI		11/30/83	
A-40	Seismic Design Criteria - Short Term Program	-	NRR/DST/GIB	USI		11/30/83	
A-41	Long Term Seismic Program	Colmar	NRR/DE/MEB	NOTE 3(b)	1	12/31/84	NA
A-42	Pipe Cracks in Boiling Water Reactors	-	NRR/DST/GIB	USI		11/30/83	B-05
A-43	Containment Emergency Sump Performance	-	NRR/DST/GIB	USI		11/30/83	
A-44	Station Blackout	-	NRR/DST/GIB	USI		11/30/83	
A-45	Shutdown Decay Heat Removal Requirements	-	NRR/DST/GIB	USI		11/30/83	
A-46	Seismic Qualification of Equipment in Operating Plants	-	NRR/DST/GIB	USI		11/30/83	
A-47	Safety Implications of Control Systems	-	NRR/DST/GIB	USI		11/30/83	
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	-	NRR/DST/GIB	USI		11/30/83	
A-49	Pressurized Thermal Shock	-	NRR/DST/GIB	USI		11/30/83	
B-1	Environmental Technical Specifications	-	NRR/DE/EHEB	E (NOTE 3)		11/30/83	NA
B-2	Forecasting Electricity Demand	-	NRR	E (NOTE 3)		11/30/83	NA
B-3	Event Categorization	-	NRR/DSI/RSB	LI (DROP)		11/30/83	NA
B-4	ECCS Reliability	Emrit	NRR/DSI/RSB	II.E.3.2		11/30/83	NA
B-5	Ductility of Two Way Slabs and Shells and Buckling Behavior of Steel Containments	Thatcher	NRR/DE/SGEB	MEDIUM		11/30/83	
B-6	Loads, Load Combinations, Stress Limits	Pittman	NRR/DE/MEB	HIGH		11/30/83	
B-7	Secondary Accident Consequence Modeling	-	NRR/DSI/AEB	LI (DROP)		11/30/83	NA
B-8	Locking Out of ECCS Power Operated Valves	Riggs	NRR/DSI/RSB	DROP		11/30/83	NA
B-9	Electrical Cable Penetrations of Containment	Emrit	NRR/DSI/PSB	NOTE 3(b)		11/30/83	NA
B-10	Behavior of BWR Mark III Containments	V'Molen	NRR/DSI/CSB	NOTE 3(a)	1	12/31/84	NA
B-11	Subcompartment Standard Problems	-	NRR/DSI/CSB	LI		11/30/83	NA

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Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
B-12	Containment Cooling Requirements (Non-LOCA)	Emrit	NRR/DSI/CSB	NOTE 3(a)		11/30/83	
B-13	Marviken Test Data Evaluation	-	NRR/DSI/CSB	LI		11/30/83	NA
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA	Emrit	NRR/DST/GIB	A-48		11/30/83	NA
B-15	CONTEMPT Computer Code Maintenance	-	NRR/DSI/CSB	LI (DROP)		11/30/83	NA
B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	Emrit	NRR/DE/MEB	A-18		11/30/83	NA
B-17	Criteria for Safety Related Operator Actions	Milstead	NRR/DHFS/LQB	MEDIUM		11/30/83	
B-18	Vortex Suppression Requirements for Containment Sumps	Emrit	NRR/DST/GIB	A-43		11/30/83	NA
B-19	Thermal-Hydraulic Stability	Colmar	NRR/DSI/CPB	NOTE 4		11/30/83	
B-20	Standard Problem Analysis	-	RES/DAE/AMBR	LI		11/30/83	
B-21	Core Physics	-	NRR/DSI/CPB	LI (DROP)		11/30/83	NA
B-22	LWR Fuel	V'Molen	NRR/DSI/CPB	NOTE 4		11/30/83	
B-23	LMFBR Fuel	-	NRR/DSI/CPB	LI (DROP)		11/30/83	NA
B-24	Seismic Qualification of Electrical and Mechanical Components	Emrit	NRR	A-46		11/30/83	NA
B-25	Piping Benchmark Problems	-	NRR/DE/MEB	LI		11/30/83	
B-26	Structural Integrity of Containment Penetrations	Riggs	NRR/DE/MTEB	NOTE 3(b)	1	12/31/84	NA
B-27	Implementation and Use of Subsection NF	-	NRR/DE/MEB	LI		11/30/83	
B-28	Radionuclide/Sediment Transport Program	-	NRR/DE/EHEB	E (NOTE 3)		11/30/83	NA
B-29	Effectiveness of Ultimate Heat Sinks	Pittman	NRR/DE/EHEB	NOTE 4		11/30/83	
B-30	Design Basis Floods and Probability	-	NRR/DE/EHEB	LI		11/30/83	
B-31	Dam Failure Model	Milstead	NRR/DE/SGEB	NOTE 4		11/30/83	
B-32	Ice Effects on Safety Related Water Supplies	Pittman	NRR/DE/EHEB	NOTE 4		11/30/83	
B-33	Dose Assessment Methodology	-	NRR/DSI/RAB	LI (NOTE 3)		11/30/83	NA
B-34	Occupational Radiation Exposure Reduction	Emrit	NRR/DSI/RAB	III.D.3.1		11/30/83	NA
B-35	Confirmation of Appendix I Models for Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light Water Cooled Power Reactors	-	NRR/DSI/METB	LI		11/30/83	
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems	Emrit	NRR/DSI/METB	NOTE 3(a)		11/30/83	
B-37	Chemical Discharges to Receiving Waters	-	NRR/DE/EHEB	E		11/30/83	
B-38	Reconnaissance Level Investigations	-	NRR/DE/EHEB	E (DROP)		11/30/83	NA
B-39	Transmission Lines	-	NRR/DE/EHEB	E (DROP)		11/30/83	NA
B-40	Effects of Power Plant Entrainment on Plankton	-	NRR/DE/EHEB	E (DROP)		11/30/83	NA
B-41	Impacts on Fisheries	-	NRR/DE/EHEB	E (DROP)		11/30/83	NA
B-42	Socioeconomic Environmental Impacts	-	NRR/DE/SAB	E (NOTE 3)		11/30/83	NA
B-43	Value of Aerial Photographs for Site Evaluation	-	NRR/DE/EHEB	E		11/30/83	
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants	-	NRR/DE/SAB	E (NOTE 3)		11/30/83	NA
B-45	Need for Power - Energy Conservation	-	NRR/DE/SAB	E (B-2)		11/30/83	NA
B-46	Cost of Alternatives in Environmental Design	-	NRR/DE/SAB	E (DROP)		11/30/83	NA

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Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
B-47	Inservice Inspection of Supports-Classes 1, 2, 3, and MC Components	Colmar	NRR/DE/MTEB	DROP		11/30/83	NA
B-48	BWR CRD Mechanical Failure (Collet Housing)	Emrit	NRR/DE/MTEB	NOTE 3(b)		11/30/83	
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments	-	NRR	LI		11/30/83	
B-50	Post-Operating Basis Earthquake Inspection	Colmar	NRR/DE/SGEB	NOTE 4		11/30/83	
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components	Emrit	NRR/DE/MEB	A-40		11/30/83	NA
B-52	Fuel Assembly Seismic and LOCA Responses	Emrit	NRR/DST/GIB	A-2		11/30/83	NA
B-53	Load Break Switch	Sege	NRR/DSI/PSB	RI (NOTE 3)		11/30/83	
B-54	Ice Condenser Containments	Milstead	NRR/DSI/CSB	NOTE 3(b)	1	12/31/84	NA
B-55	Improved Reliability of Target Rock Safety-Relief Valves	V'Molen	NRR/DE/MEB	MEDIUM		11/30/83	
B-56	Diesel Reliability	Milstead	NRR/DSI/PSB	HIGH		11/30/83	
B-57	Station Blackout	Emrit	NRR/DST/GIB	A-44		11/30/83	
B-58	Passive Mechanical Failures	Colmar	NRR/DE/EQB	MEDIUM		11/30/83	
B-59	N-1 Loop Operation in BWRs and PWRs	Colmar	NRR/DSI/RSB	NOTE 4		11/30/83	
B-60	Loose Parts Monitoring System	Emrit	NRR/DSI/CPB	NOTE 3(b)	1	12/31/84	NA
B-61	Allowable ECCS Equipment Outage Periods	Pittman	NRR/DST/RRAB	MEDIUM		11/30/83	
B-62	Reexamination of Technical Bases for Establishing SLs, LSSSs, and Reactor Protection System Trip Functions	-	NRR/DSI/CPB	LI (DROP)		11/30/83	NA
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary	Emrit	NRR/DE/MEB	NOTE 3(a)		11/30/83	
B-64	Decommissioning of Reactors	Colmar	NRR/DE/CHEB	NOTE 2		11/30/83	
B-65	Iodine Spiking	Milstead	NRR/DSI/AEB	DROP	2	12/31/84	NA
B-66	Control Room Infiltration Measurements	Matthews	NRR/DSI/AEB	NOTE 3(a)		11/30/83	
B-67	Effluent and Process Monitoring Instrumentation	Colmar	NRR/DSI/METB	III.D.2.1		11/30/83	NA
B-68	Pump Overspeed During LOCA	Riani	NRR/DSI/ASB	DROP		11/30/83	NA
B-69	ECCS Leakage Ex-Containment	Riani	NRR/DSI/METB	III.D.1.1		11/30/83	NA
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps	Emrit	NRR/DSI/PSB	NOTE 3(a)		11/30/83	
B-71	Incident Response	Riani	NRR	III.A.3.1		11/30/83	NA
B-72	Health Effects and Life Shortening from Uranium and Coal Fuel Cycles	-	NRR/DSI/RAB	LI		11/30/83	NA
B-73	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel	Thatcher	NRR/DE/MEB	C-12		11/30/83	NA
C-1	Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	Milstead	NRR/DE/EQB	NOTE 3(a)		11/30/83	
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure	Emrit	NRR/DSI/CSB	NOTE 3(b)		11/30/83	NA
C-3	Insulation Usage Within Containment	Emrit	NRR/DST/GIB	A-43		11/30/83	NA
C-4	Statistical Methods for ECCS Analysis	Riggs	NRR/DSI/RSB	NOTE 4		11/30/83	
C-5	Decay Heat Update	Riggs	NRR/DSI/CPB	NOTE 4		11/30/83	

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
C-6	LOCA Heat Sources	Riggs	NRR/DSI/CPB	NOTE 4		11/30/83	
C-7	PWR System Piping	Emrit	NRR/DE/MTEB	NOTE 3(b)		11/30/83	NA
C-8	Main Steam Line Leakage Control Systems	Milstead	NRR/DSI/ASB	HIGH		11/30/83	
C-9	RHR Heat Exchanger Tube Failures	V'Molen	NRR/DSI/RSB	DROP		11/30/83	NA
C-10	Effective Operation of Containment Sprays in a LOCA	Emrit	NRR/DSI/AEB	NOTE 3(a)		11/30/83	NA
C-11	Assessment of Failure and Reliability of Pumps and Valves	Matthews	NRR/DE/MEB	MEDIUM		11/30/83	
C-12	Primary System Vibration Assessment	Thatcher	NRR/DE/MEB	NOTE 3(b)		11/30/83	NA
C-13	Non-Random Failures	Emrit	NRR/DST/GIB	A-17		11/30/83	NA
C-14	Storm Surge Model for Coastal Sites	Emrit	NRR/DE/EHEB	NOTE 4		11/30/83	
C-15	NUREG Report for Liquids Tank Failure Analysis	-	NRR/DE/EHEB	LI (DROP)		11/30/83	NA
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection	-	NRR/DE/EHEB	E (DROP)		11/30/83	NA
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	Emrit	NRR/DSI/METB	NOTE 3(a)		11/30/83	NA
D-1	Advisability of a Seismic Scram	Thatcher	RES/DET/MSEB	LOW		11/30/83	NA
D-2	Emergency Core Cooling System Capability for Future Plants	Emrit	NRR/DSI/RSB	NOTE 4		11/30/83	
D-3	Control Rod Drop Accident	Emrit	NRR/DSI/CPB	NOTE 3(b)		11/30/83	NA

NEW GENERIC ISSUES

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
1.	Failures in Air-Monitoring, Air-Cleaning, and Ventilating Systems	Emrit	NRR/DSI/METB	DROP		11/30/83	NA
2.	Failure of Protective Devices on Essential Equipment	Colmar	NRR/DSI/ICSB	NOTE 4		11/30/83	
3.	Set Point Drift in Instrumentation	Emrit	NRR/DSI/ICSB	NOTE 2		11/30/83	
4.	End-of-Life and Maintenance Criteria	Thatcher	NRR/DE/EQB	NOTE 3(b)		11/30/83	NA
5.	Design Check and Audit of Balance-of-Plant Equipment	Pittman	NRR/DSI/ASB	I.F.1		11/30/83	NA
6.	Separation of Control Rod from Its Drive and BWR High Rod Worth Events	V'Molen	NRR/DSI/CPB	NOTE 3(b)		11/30/83	NA
7.	Failures Due to Flow-Induced Vibrations	V'Molen	NRR/DSI/RSB	DROP		11/30/83	NA
8.	Inadvertent Actuation of Safety Injection in PWRs	Colmar	NRR/DSI/RSB	I.C.1		11/30/83	NA
9.	Reevaluation of Reactor Coolant Pump Trip Criteria	Emrit	NRR/DSI/RSB	II.K.3		11/30/83	NA
10.	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges	Riggs	NRR/DSI/ICSB	DROP		11/30/83	NA
11.	Turbine Disc Cracking	Pittman	NRR/DE/MTEB	A-37		11/30/83	NA
12.	BWR Jet Pump Integrity	Sege	NRR/DE/MTEB, MEB	NOTE 3(b)	1	12/31/84	NA
13.	Small Break LOCA from Extended Overheating of Pressurizer Heaters	Riani	NRR/DSI/RSB	DROP		11/30/83	NA
14.	PWR Pipe Cracks	Matthews	NRR/DE/MTEB	NOTE 2		11/30/83	
15.	Radiation Effects on Reactor Vessel Supports	Emrit	NRR/DE/MTEB	LOW		11/30/83	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
16.	BWR Main Steam Isolation Valve Leakage Control Systems	Milstead	NRR/DSI/ASB	C-8		11/30/83	NA
17.	Loss of Offsite Power Subsequent to LOCA	Colmar	NRR/DSI/PSB, ICSB	DROP		11/30/83	NA
18.	Steam Line Break with Consequential Small LOCA	Riggs	NRR/DSI/RSB	I.C.1		11/30/83	NA
19.	Safety Implications of Nonsafety Instrument and Control Power Supply Bus	Sege	NRR/DST/GIB	A-47		11/30/83	NA
20.	Effects of Electromagnetic Pulse on Nuclear Plant Systems	Thatcher	NRR/DSI/ICSB	NOTE 3(b)	1	6/30/84	NA
21.	Vibration Qualification of Equipment	Thatcher	NRR/DE/EQB	NOTE 4		11/30/83	
22.	Inadvertent Boron Dilution Events	V'Molen	NRR/DSI/RSB	NOTE 3(b)	1	12/31/84	NA
23.	Reactor Coolant Pump Seal Failures	Riggs	NRR/DSI/ASB	HIGH		11/30/83	
24.	Automatic Emergency Core Cooling System Switch to Recirculation	V'Molen	NRR/DSI/RSB	NOTE 4		11/30/83	
25.	Automatic Air Header Dump on BWR Scram System	Milstead	NRR/DSI/RSB	NOTE 3(a)		11/30/83	
26.	Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power	Emrit	NRR/DSI/ASB	17		11/30/83	NA
27.	Manual vs. Automated Actions	Pittman	NRR/DSI/RSB	B-17		11/30/83	NA
28.	Pressurized Thermal Shock	Emrit	NRR/DST/GIB	A-49		11/30/83	NA
29.	Bolting Degradation or Failure in Nuclear Power Plants	V'Molen	NRR/DE/MTEB	HIGH		11/30/83	
30.	Potential Generator Missiles - Generator Rotor Retaining Rings	Pittman	NRR	NOTE 4		11/30/83	
31.	Natural Circulation Cooldown	Riggs	NRR/DSI/RSB	I.C.1		11/30/83	NA
32.	Flow Blockage in Essential Equipment Caused by Corbicula	Emrit	NRR/DSI/ASB	51		11/30/83	NA
33.	Correcting Atmospheric Dump Valve Opening Upon Loss of Integrated Control System Power	Pittman	NRR/DSI/ICSB	A-47		11/30/83	NA
34.	RCS Leak	Riggs	NRR/DHFS/PSRB	DROP	1	6/30/84	NA
35.	Degradation of Internal Appurtenances in LWRs	V'Molen	NRR/DSI/CPB, RSB	LOW		11/30/83	NA
36.	Loss of Service Water	Colmar	NRR/DSI/ASB, AEB, RSB	NOTE 1	1	6/30/84	
37.	Steam Generator Overfill and Combined Primary and Secondary Blowdown	Colmar	NRR	NOTE 4		11/30/83	
38.	Potential Recirculation System Failure as a Consequence of Injection of Containment Paint Flakes or Other Fine Debris	Milstead	NRR	NOTE 4		11/30/83	
39.	Potential for Unacceptable Interaction Between the CRD System and Non-Essential Control Air System	Pittman	NRR/DSI/ASB	25		11/30/83	NA
40.	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	Colmar	NRR/DSI/ASB	NOTE 3(a)	1	6/30/84	B-65
41.	BWR Scram Discharge Volume Systems	V'Molen	NRR/DSI/RSB	NOTE 3(a)		11/30/83	B-58
42.	Combination Primary/Secondary System LOCA	Riggs	NRR/DSI/RSB	18		11/30/83	NA
43.	Contamination of Instrument Air Lines	Milstead	NRR/DSI/ASB	DROP		11/30/83	NA
44.	Failure of Saltwater Cooling System	Milstead	NRR/DSI/ASB	43		11/30/83	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
45.	Inoperability of Instrumentation Due to Extreme Cold Weather	Milstead	NRR/DSI/ICSB	NOTE 3(a)	1	6/30/84	
46.	Loss of 125 Volt DC Bus	Sege	NRR/DSI/PSB	76		11/30/83	NA
47.	Loss of Off-Site Power	Thatcher	NRR/DSI/RSB, ASB	NOTE 3(b)		11/30/83	
48.	LCO for Class 1E Vital Instrument Buses in Operating Reactors	Sege	NRR/DSI/PSB	NOTE 2		11/30/83	
49.	Interlocks and LCOs for Redundant Class 1E Tie Breakers	Sege	NRR/DSI/PSB	MEDIUM	1	12/31/84	
50.	Reactor Vessel Level Instrumentation in BWRs	Thatcher	NRR/DSI/RSB, ICSB	NOTE 3(b)	1	12/31/84	NA
51.	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	Emrit	NRR/DSI/ASB	MEDIUM		11/30/83	
52.	SSW Flow Blockage by Blue Mussels	Emrit	NRR/DSI/ASB	51		11/30/83	NA
53.	Consequences of a Postulated Flow Blockage Incident in a BWR	V'Molen	NRR/DSI/CPB, RSB	DROP	1	12/31/84	NA
54.	Valve Operator-Related Events Occurring During 1978, 1979, and 1980	Colmar	NRR	NOTE 4		11/30/83	
55.	Failure of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand	Emrit	NRR/DSI/PSB	NOTE 4		11/30/83	
56.	Abnormal Transient Operating Guidelines as Applied to a Steam Generator Overfill Event	Colmar	NRR/DHFS/HFEB	A-45, I.D.1		11/30/83	NA
57.	Effects of Fire Protection System Actuation on Safety-Related Equipment	Matthews	NRR	NOTE 4		11/30/83	
58.	Inadvertent Containment Flooding	Sege	NRR/DSI/ASB, CSB	DROP		11/30/83	
59.	Technical Specification Requirements for Plant Shutdown when Equipment for Safe Shutdown is Degraded or Inoperable	Emrit	NRR	NOTE 4		11/30/83	
60.	Lamellar Tearing of Reactor Systems Structural Supports	Colmar	NRR/DST/GIB	A-12		11/30/83	NA
61.	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments	Milstead	NRR/DSI/CSB	MEDIUM		11/30/83	
62.	Reactor Systems Bolting Applications	V'Molen	NRR	NOTE 4		11/30/83	
63.	Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis	V'Molen	NRR	NOTE 4		11/30/83	
64.	Identification of Protection System Instrument Sensing Lines	Thatcher	NRR/DSI/ICSB	NOTE 3(b)		11/30/83	
65.	Probability of Core-Melt Due to Component Cooling Water System Failures	V'Molen	NRR/DSI/ASB	HIGH		11/30/83	
66.	Steam Generator Requirements	Riggs	NRR/DL/ORAB	NOTE 2		11/30/83	
67.	Steam Generator Staff Actions	Riggs	NRR	NOTE 4		11/30/83	
68.	Postulated Loss of Auxiliary Feedwater System Resulting from Turbine-Driven Auxiliary Feedwater Pump Steam Supply Line Rupture	Pittman	NRR/DSI/ASB	HIGH	1	6/30/84	
69.	Make-up Nozzle Cracking in B&W Plants	Colmar	NRR/DE/MEB, MTEB	NOTE 3(b)	1	12/31/84	(later)
70.	PORV and Block Valve Reliability	Riggs	NRR/DSI/RSB	MEDIUM	1	6/30/84	

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
71.	Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety	Matthews	NRR	NOTE 4		11/30/83	
72.	Control Rod Drive Guide Tube Support Pin Failures	V'Molen	NRR	NOTE 4		11/30/83	
73.	Detached Thermal Sleeves	Colmar	NRR	NOTE 4		11/30/83	
74.	Reactor Coolant Activity Limits for Operating Reactors	Milstead	NRR	NOTE 4		11/30/83	
75.	Generic Implications of ATWS Events at the Salem Nuclear Plant	Thatcher	NRR/DSI	NOTE 1		11/30/83	
76.	Instrumentation and Control Power Interactions	Colmar	NRR	NOTE 4		11/30/83	
77.	Flooding of Safety Equipment Compartments by Back-flow Through Floor Drains	Colmar	NRR/DSI/ASB	HIGH		11/30/83	
78.	Monitoring of Fatigue Transient Limits for Reactor Coolant System	Riggs	NRR	NOTE 4		11/30/83	
79.	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	Colmar	NRR/DE/MEB, NRR/DSI/RSB	MEDIUM	1	12/31/84	
80.	Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments	V'Molen	NRR/DSI/RSB, ASB, CPB	LOW		11/30/83	NA
81.	Impact of Locked Doors and Barriers on Plant Personnel and Safety	Colmar	NRR/DHFS/PSRB	DROP	1	12/31/84	NA
82.	Beyond Design Basis Accidents in Spent Fuel Pools	V'Molen	NRR/DSI/AEB	MEDIUM		11/30/83	
83.	Control Room Habitability	Matthews	NRR	NOTE 4		11/30/83	
84.	CE PORVs	Riggs	NRR	NOTE 4		11/30/83	
85.	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments	Milstead	NRR	NOTE 4		11/30/83	
86.	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	Emrit	NRR/DE/MTEB	NOTE 2		12/31/84	
87.	Failure of HPCI Steam Line Without Isolation	Pittman	NRR	NOTE 4		(later)	
88.	Earthquakes and Emergency Planning	Riggs	NRR	NOTE 4		(later)	
89.	Stiff Pipe Clamps	Riggs	NRR	NOTE 4		(later)	
90.	Technical Specifications for Anticipatory Trips	V'Molen	NRR/DSI/RSB, ICSB	LOW		12/31/84	NA
91.	Main Crankshaft Failures in Transamerica DeLaval Emergency Diesel Generators	Emrit	NRR	NOTE 4		(later)	
92.	Fuel Crumbling During LOCA	V'Molen	NRR/DSI/RSB, CPB	LOW		12/31/84	NA
93.	Steam Binding of Auxiliary Feedwater Pumps	Pittman	NRR/DSI/ASB	HIGH		12/31/84	
94.	Additional Low Temperature Overpressure Protection Issues for Light Water Reactors	Pittman	NRR	NOTE 4		(later)	
95.	Loss of Effective Volume for Containment Recirculation Spray	Milstead	NRR	NOTE 4		(later)	
96.	RHR Suction Valve Testing	V'Molen	NRR	NOTE 4		(later)	
97.	PWR Reactor Cavity Uncontrolled Exposures	V'Molen	NRR	NOTE 4		(later)	
98.	CRD Accumulator Check Valve Leakage	Pittman	NRR	NOTE 4		(later)	
99.	RCS/RHR Suction Line Valve Interlock on PWRs	Pittman	NRR	NOTE 4		(later)	
100.	OTSG Level	Pittman	NRR	NOTE 4		(later)	
101.	Break Plus Single Failure in BWR Water Level Instrumentation	V'Molen	NRR	NOTE 4		(later)	
102.	Human Error in Events Involving Wrong Unit or Wrong Train	Emrit	NRR	NOTE 4		(later)	

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
103.	Design for Probable Maximum Precipitation	Emrit	NRR	NOTE 4		(later)	
104.	Reduction of Boron Dilution Requirements	V'Molen	NRR	NOTE 4		(later)	
105.	Interfacing System LOCA at BWRs	Milstead	NRR	NOTE 4		(later)	
106.	Piping and Use of Highly Combustible Gases in Vital Areas	Colmar	NRR	NOTE 4		(later)	
107.	Generic Implications of Main Transformer Failures	Colmar	NRR	NOTE 4		(later)	
108.	BWR Suppression Pool Temperature Limits	Colmar	NRR	NOTE 4		(later)	
109.	Reactor Vessel Closure Failure	Pittman	NRR	NOTE 4		(later)	
110.	Equipment Protective Devices on Engineered Safety Features	Pittman	NRR	NOTE 4		(later)	
	HUMAN FACTORS ISSUES						
HF01	HUMAN FACTORS PROGRAM PLAN (HFPP)						
HF01.1.0	Staffing and Qualifications	-	-	-			
HF01.1.1	NPP Staffing Requirements	Pittman	NRR/DHFS/LQB	HIGH		12/31/84	
HF01.1.2	NPP Personnel Qualifications Requirements	Pittman	NRR/DHFS/LQB	HIGH		12/31/84	
HF01.1.3	Guidance on Limits and Conditions of Shift Work	Pittman	NRR/DHFS/LQB	HIGH		12/31/84	
HF01.1.4	Fitness for Duty	Pittman	NRR/DHFS/LQB	HIGH		12/31/84	
HF01.2.0	Training	-	-	-			
HF01.2.1	Development of Training Regulation and Guidance	Pittman	NRR/DHFS/LQB	HIGH		12/31/84	
HF01.2.2	NRC Training Evaluation Program	Pittman	NRR/DHFS/LQB	HIGH		12/31/84	
HF01.3.0	Licensing Examination	-	-	-			
HF01.3.1	The Examination Content	Pittman	NRR/DHFS/OLB	HIGH		12/31/84	
HF01.3.2	The Examination Process	Pittman	NRR/DHFS/OLB	HIGH		12/31/84	
HF01.4.0	Procedures	-	-	-			
HF01.4.1	Procedures Guidance and Criteria	Pittman	NRR/DHFS/PSRB	HIGH		12/31/84	
HF01.5.0	Man-Machine Interface (MMI)	-	-	-			
HF01.5.1	MMI Guidance for Existing Designs	Pittman	NRR/DHFS/HFEB	HIGH		12/31/84	
HF01.5.2	MMI Guidance for Designs Based on Advanced Technologies	Pittman	NRR/DHFS/HFEB	HIGH		12/31/84	
HF01.6.0	Management and Organization	-	-	-			
HF01.6.1	Regulatory Position on Management and Organization at Operating Reactors	Pittman	NRR/DHFS/LQB	HIGH		12/31/84	
HF01.6.2	NRC Management and Organization Guidelines and Assessment Procedures for Operating License Reviews	Pittman	NRR/DHFS/LQB	HIGH		12/31/84	

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
HF01.7.0	Human Reliability	-	-	-			
HF01.7.1	Human Error Data Acquisition	Pittman	RES	LI		12/31/84	
HF01.7.2	Human Error Data Storage and Retrieval	Pittman	RES	LI		12/31/84	
HF01.7.3	Reliability Evaluation Specialist Aids	Pittman	RES	LI		12/31/84	
HF01.7.4	Safety Event Analysis Results Application	Pittman	RES	LI		12/31/84	
HF02	MAINTENANCE AND SURVEILLANCE PROGRAM						
HF02.1	Maintenance and Surveillance Program	Pittman	NRR/DHFS/LQB	Note 4		(later)	

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TABLE III

SUMMARY OF THE PRIORITIZATION OF ALL TMI ACTION PLAN ITEMS, TASK ACTION PLAN ITEMS, NEW GENERIC ISSUES, AND HUMAN FACTORS ISSUES

Legend

NOTES:
1 - Possible Resolution Identified for Evaluation
2 - Resolution Available
3 - Resolution Resulted in either the Establishment of New Requirements or No New Requirements
4 - Issues to be Prioritized in the Future
5 - Issue that is not a Generic Safety Issue but should be Assigned Resources for Completion

HIGH	- High Safety Priority
MEDIUM	- Medium Safety Priority
LOW	- Low Safety Priority
DROP	- Issue Dropped as a Generic Issue
USI	- Unresolved Safety Issue
I	- TMI Action Plan Item with Implementation of Resolution Mandated by NUREG-0737

TABLE III (Continued)

ACTION ITEM/ISSUE GROUP	I	COVERED IN OTHER ISSUES	RESOLVED STAGES NOTE 1	NOTE 2	NOTE 3	USI	HIGH	MEDIUM	LOW	DROP	NOTE 4	NOTE 5	TOTAL
1. TMI ACTION PLAN ITEMS (352)													
(a) Safety													
(i) Generic Safety	94	61	6	1	89	0	12	6	13	7	2	-	291
(b) Non-Safety													
(i) Licensing	-	0	0	4	51	-	-	-	-	0	0	6	61
2. TASK ACTION PLAN ITEMS (142)													
(a) Safety													
(i) Generic Safety	-	17	0	1	25	27	4	7	3	9	13	-	106
(ii) Regulatory Impact	-	0	0	0	1	-	-	-	-	0	0	1	2
(b) Non-Safety													
(i) Licensing	-	0	0	0	1	-	-	-	-	7	0	11	19
(ii) Environmental	-	1	0	0	6	-	-	-	-	6	0	2	15
3. NEW GENERIC ISSUES (110)													
(a) Safety													
(i) Generic Safety	-	20	2	5	13	0	6	6	5	10	43	-	110
4. HUMAN FACTORS ISSUES (18)													
(a) Safety													
(i) Generic Safety	-	0	0	0	0	0	13	0	0	0	1	-	14
(b) Non-Safety													
(i) Licensing	-	0	0	0	0	-	-	-	-	0	0	4	4
TOTAL:	94	99	8	11	186	27	35	19	21	39	59	24	622

TABLE IV

LISTING OF AEOD REPORTS AND RELATED GENERIC ISSUES

This listing shows all AEOD reports that have been addressed either as completely new safety issues or as part of new or existing safety issues. It should be noted that, in some cases, more than one AEOD report has been generated on a single topic. However, all AEOD reports related to the identified safety issues are listed alphanumerically including those that have been superseded by other AEOD reports. The following is a description of the types of AEOD reports:

C - Reactor Case Study
E - Reactor Engineering Evaluation
S - Special Study Report
T - Technical Review Report

AEOD Report No.	AEOD Report Title	Related Safety Issue No.	Related AEOD Report
C001	Report on the Browns Ferry 3 Partial Failure to Scram Event on June 28, 1980	41	-
C003	Report on Loss of Offsite Power Event at Arkansas Nuclear One, Units 1 and 2	47	-
C004	AEOD Actions Concerning the Crystal River 3 Loss of Non-Nuclear Instrumentation and Integrated Control System Power on February 26, 1980	33	E122
C005	AEOD Observations and Recommendations Concerning the Problem of Steam Generator Overfill and Combined Primary and Secondary Side Blowdown	37, 42	-
C101	Report on the Saint Lucie 1 Natural Circulation Cooldown on June 11, 1980	31	-
C102	H. B. Robinson Reactor Coolant System Leak on January 29, 1981	34	-
C103	AEOD Safety Concerns Associated with Pipe Breaks in the BWR Scram System	40	-
C104	Millstone Unit 2 Loss of 125 V DC Bus Event on January 2, 1981	46	-
C105	Report on the Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980	36	-
C201	Safety Concern Associated with Reactor Vessel Level Instrumentation in Boiling Water Reactors	50, 101	-

TABLE IV (Continued)

AEOD Report No.	AEOD Report Title	Related Safety Issue No.	Related AEOD Report
C202	Report on Service Water System Flow Blockages by Bivalve Mollusks at Arkansas Nuclear One and Brunswick	32	E016
C203	Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980	54	-
C204	San Onofre Unit 1 Loss of Salt Water Cooling Event of March 10, 1980	44	-
C205	Abnormal Transient Operating Guidelines (ATOG) as Applied to the April 1981 Overfill Event at Arkansas Nuclear One, Unit 1	56	-
C301	Failures of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand	55	-
C401	Low Temperature Overpressure Events at Turkey Point Unit 4	94	E426
C404	Steam Binding of Auxiliary Feedwater Pumps	93	E325
E002	BWR Jet Pump Integrity	12	-
E005	Operational Restrictions for Class 1E 120 VAC Vital Instrument Buses	48	-
E007	Potential for Unacceptable Interaction Between the Control Rod Drive System and Non-Essential Control Air System at the Browns Ferry Plant	39	-
E010	Tie Breaker Between Redundant Class 1E Buses - Point Beach Nuclear Plant, Units 1 and 2	49	-
E011	Concerns Relating to the Integrity of a Polymer Coating for Surfaces Inside Containment	38	-
E016	Flow Blockage in Essential Equipment at ANO Caused by <u>Corbicula</u> sp. (Asiatic Clams)	32	C202
E101	Degradation of Internal Appurtenances in LWR Piping	35	-
E112	Inoperability of Instrumentation Due to Extreme Cold Weather	45	E226
E122	AEOD Concern Regarding Inadvertent Opening of Atmospheric Dump Valves on B&W Plants During Loss of ICS/NNI Power	33	C004
E123	Common Cause Failure Potential at Rancho Seco - Desiccant Contamination of Air Lines	43	-
E204	Effects of Fire Protection System Actuation on Safety-Related Equipment	57	-
E209	Generator Rotor Retaining Ring as a Potential Missile (Incident at Barseback 1 on 4/13/79)	30	-
E215	Engineering Evaluation of the Salt Service Water System Flow Blockage at the Pilgrim Nuclear Power Station by Blue Mussels	52	-
E226	Inoperability of Instrumentation Due to Extreme Cold Weather	45	E112

TABLE IV (Continued)

AEOD Report No.	AEOD Report Title	Related Safety Issue No.	Related AEOD Report
E304	Investigation of Backflow Protection in Common Equipment and Floor Drain Systems to Prevent Flooding of Vital Equipment in Safety-Related Compartments	77	-
E325	Vapor Binding of Auxiliary Feedwater Pumps at Robinson 2	93	C404
E414	Stuck Open Isolation Check Valve on the Residual Heat Removal System at Hatch Unit 2	105	-
E417	Loosening of Flange Bolts on RHR Heat Exchanger Leading to Primary to Secondary Side Leakage	C-9	-
E426	Single Failure Vulnerability of Power Operated Relief Valve (PORV) Actuation Circuitry for Low Temperature Overpressure Protection (LTOP)	94	C401
S401	Human Error in Events Involving Wrong Unit or Wrong Train	102	-
T302	Postulated Loss of Auxiliary Feedwater System Resulting from a Turbine Driven Auxiliary Feedwater Pump Steam Supply Line Rupture	68	-
T305	Flow Blockage in Essential Raw Cooling Water System Due to Asiatic Clam Instrusion at Sequoyah 1	51	-

TASK I.A.2: TRAINING AND QUALIFICATIONS OF OPERATING PERSONNEL

The objectives of this task are as follows: (1) to improve the capability of operators and supervisors to understand and control complex reactor transients and accidents, (2) to improve the general capability of an operations organization to respond rapidly and effectively to upset conditions, and (3) to increase the education, experience, and training requirements for operators, senior operators, supervisors, and other personnel in the operations organization to substantially improve their capability to perform their duties.

ITEM I.A.2.2: TRAINING AND QUALIFICATIONS OF OPERATIONS PERSONNEL

DESCRIPTION

Under the TMI Action Plan,[48] the NRC may require reactor licensees to review their training and qualification programs for all operations personnel. This is interpreted to include licensed and auxiliary operators, technicians, maintenance personnel and supervisors. The review is to examine current practices in light of the safety significance of the duties of the operations staff. If the review determines that the current practices adequately assure proper safety-related staff conduct, then documentation of the justification for this determination is required. The documentation need not be submitted to the NRC but must be maintained on site. If the review uncovers inadequacies, the licensee is required to upgrade the training and qualification practices to ensure adequate performance of operations personnel. The evaluation of this issue includes the consideration of Item I.A.2.6(3).

PRIORITY DETERMINATION

The first step in estimating the effect of training reviews on operator-error contributions to plant risk was to assemble a panel of experts from the PNL staff. This panel represented considerable experience in reactor operations, utility training programs, and reactor plant systems. The panel included members with utility field experience and reactor operator licensing examiners.

The judgments of the panel, as detailed below, are based on the two following considerations:[64]

(1) The potential effect of this issue is limited by its semi-voluntary nature. That is, the judgment of adequacy is in the hands of the individual utilities. Furthermore, the current INPO and NRC research work in task analysis deals with generic routine operations. Plant-specific operation and operation under upset conditions are left to the individual utilities. This dilutes the effectiveness of the task analysis efforts in providing the basis for the training and qualification review.

Related issues which are supported by and in turn support this issue are the conducting of plant drills and accreditation of training programs.

While neither of these is directly required by the training and qualifications review, both could be a part of the response and both would have a positive effect on personnel performance.

(2) There is a wide variation among utilities in both the training programs and the performance of operations staff. In many facilities there is much room for improvement. Therefore, while the potential effect of the training and qualifications review effort is limited, a significant overall reduction in safety-related human error for operations personnel is expected because of the wide margin available for improvement.

Assumptions

In estimating the benefit and costs, the PNL panel divided licensees into three groups.

(1) Minimally affected group: These utilities currently have a good effective training and qualification program and good operations personnel performance. They should be minimally affected by this safety issue. The fractional population of this group is estimated to be 15% of the reactor licensees.

(2) Intermediately affected group: These utilities' training and qualification programs and/or operations performance have room for improvement. This group, estimated to be 60% of the population, would undergo improvements and therefore be affected by the issue.

(3) Maximally affected group: These utilities have deficiencies in their training and qualification programs and in operations personnel performance. They would be significantly affected by this safety issue and major restructuring of programs would be expected. This group is estimated to contain 25% of reactor licensees.

From the estimates for these groups, weighted composite estimates can be derived. NUREG/CR-2800[64] shows the safety benefit estimates from the panel for each of the groups and also gives the weighted averages.

The values given in NUREG/CR-2800[64] are in terms of percent changes. For inclusion into the value/impact score formulation they must be converted to other measures. The reduction in human error must be transformed into the resulting reduction in risk as measured by change in probabilistic exposure (man-rem/reactor-year). The change in annual occupational exposure must be transformed from percent improvement into man-rem per reactor-year.

The reduction in risk will be developed by examining the quantitative impact on accident event frequencies of human error rates in key scenarios. The reduction in human error will thereby be translated into a reduction in accident frequency. No additional reduction due to accident mitigation will be assumed. The values given in NUREG/CR-2800,[64] for the best estimate of improvement will be used, or 17% for operator error and 28% for maintenance.

Frequency/Consequence Estimate

This issue centers around operator and maintenance training programs to improve personnel performance. This issue relates generically to both BWRs and PWRs, and ideally the risk reduction attributable to its resolution would be estimated by selecting a representative plant of each type. However, maintenance and operator performance impact essentially accident sequences in the risk equations. To save time, the calculations were performed for one representative PWR and inferences drawn for all reactors. The Oconee 3 (a RSSMAP PWR) plant risk equations developed in NUREG/CR-1659,[54] Vol. 4 (Hatch 1981) were used for this analysis.

It will be assumed that the 17% reduction in operator error can be applied directly to elements containing an operator error frequency and the 28% reduction can be applied directly to maintenance variables. This assumption introduces some error in the maintenance contribution. This is because some maintenance operations on nuclear systems have fixed times associated with cooldown and preparation, etc., in addition to the actual hands-on time for maintenance that would be subject to improvement through training. Maintenance done properly the first time also reduces the frequency of maintenance outage and downtime for proper repairs at some future date. Thus, fixed time periods in maintenance outages are indirectly reduced over the long run with improved maintenance performance simply because the need for maintenance may be reduced except for systems that undergo preventive maintenance at set intervals.

To calculate the total public risk reduction it was assumed that issue resolution would apply to all plants existing and planned as given in NUREG/CR-2800, Appendix C.[64] This would represent a grand total of 4,000 plant-years of operation (143 plants with an average life expectancy of 28 years). Implementation of the solution would provide a reduction of 9 man-rem/plant-year. For all plants, assuming a typical midwest-type meteorology and an average population density of U.S. reactor sites of 340 people per square mile, the total public risk reduction totals 122,400 man-rem.

Cost Estimate

Industry Cost: In estimating the costs to industry of implementing and operating under the resolution of this issue the PNL panel divided the industry once again into three categories. These groups and their estimates are shown in NUREG/CR-2800.[64] The total costs to industry for implementation is the product of the number of plants and the per-plant cost, (143)($0.335M) = $48M. The total operation cost is the product of the number of plants, the average remaining life, and the plant annual cost, (143)(28)($0.16M) = $640M. The overall cost to industry is the sum of the total implementation and operational cost, $[640 + 48]M = $688M.

NRC Cost: The cost for the NRC to implement the safety issue resolution was taken from NUREG-0660.[48] This called for 1.1 person-years of NRC effort which is equivalent to $110,000. The annual NRC effort through OIE to review the justification documentation and new training programs is estimated to be one person-year. This is $100,000 per year. Over the lifetime of the completed and planned reactors this is $2.8M. Therefore, the overall cost to the NRC is the sum of the implementation and operation costs, $[0.11 + 2.8]M or $2.9M.

According to PNL estimates and calculations, the total cost for the implementation and operation of this safety issue is then [$688M + $2.9M] or approximately $691M.

Value/Impact Assessment

The public risk reduction estimated for this issue is 122,400 man-rem. The value/impact score based on this result is

$$S = \frac{122{,}400 \text{ man-rem}}{\$691M}$$

$$= 177 \text{ man-rem/\$M}$$

Other Considerations

Including the occupational dose reduction (2.4×10^5 man-rem) in the value/impact equation gives a score of 524 man-rem/$M. PNL calculated[64] the occupational risk reduction for accident-related occupational exposure to be 880 man-rem. However, it was estimated that with improved training the operational doses could be reduced by 2.4×10^5 man-rem for 143 plants over the average remaining plant lifetime.

CONCLUSION

Because of the extensive number of sequences considered to be affected by this issue, the base-case risk is high at a calculated range of from 60 to 73 man-rem/plant-year. Based on the potential reduction in public risk and ORE, this issue was determined to be high priority. However, with the publication of NUREG-0985, Revision 1,[651] this item is now covered in Sections 1.3 and 2.1 of the HFPP.

ITEM I.A.2.4: NRR PARTICIPATION IN INSPECTOR TRAINING

DESCRIPTION

Historical Background

Based on NUREG-0660,[48] NRR is required to provide supplemental instruction to the OIE inspectors by the licensing and human factors staff as an addition to the already established OIE inspector training program. The purpose of such instruction would be to focus the inspector's attention on problems associated with human factors. With such training it is expected that the inspectors would become more sensitive to such problems and hence more apt to instigate corrective action and thereby improve plant safety in this area. This would provide a means of responding to the TMI-related concern on human factors problems for plant operations staff.

Safety Significance

The principal safety benefit to be derived from NRR participation in OIE inspector training is in the improvements those inspectors will bring about because of that enhanced training. The training will increase inspector awareness in human factors and personnel-related problems. In areas such as emergency procedures reviews, routine operational practices and hardware-to-human interface deficiencies may be found by inspectors and corrected. A panel of PNL experts explored the potential significance of this issue.[64] This panel included three reactor operator license examiners, members with utility field experience, experience in training as well as general reactor safety experience.

The panel envisioned that the solution of this issue would be the addition of one week of instruction in human factors to the OIE inspector training course. The staff from NRR would participate in the instruction but would probably rely on a qualified consultant to conduct the majority of the instruction. It was assumed that the principal target of the training would be the resident inspectors. The potential effect of the training upon the OIE review of emergency procedures, plant hardware and routine practices could be significant, but the overall effect is thought to be limited because of two factors: the short exposure of the inspector to human factors training, and the indirect nature of the safety benefit. That is, a marginal improvement in inspector awareness will result in some corrective actions which would result in some safety improvement. The separation between initial action and the safety benefit complicates assessment of the effectiveness of the proposed resolution of the issue.

PNL estimated[64] a human error rate reduction of 2% for operators and maintenance personnel (operations staff assumed most likely to affect plant safety). It is important to note that this is an overall industry-wide estimate. Some isolated actions could be highly significant. The PNL estimated cost for this additional training is about $1,000.

CONCLUSION

Capabilities of inspectors could clearly be improved through the proposed training. There would be an indirect effect on risk, since better trained inspectors would identify more cost-effective improvements in plant operations. However, there is no reasonable way that the magnitude of the safety significance and cost of these improvements can be estimated quantitatively. This additional training would enhance the capabilities and thus contribute to the effectiveness and efficiency of the NRC in performing its regulatory safety mission. Thus, this training proposal should be evaluated as a Licensing Issue.

ITEM I.A.2.5: PLANT DRILLS

DESCRIPTION

The intent of this TMI Action Plan item is to upgrade operator training by requiring operating personnel to conduct plant drills during shifts. Normal and off-normal operating maneuvers would be simulated for walk-through drills on a plant-wide basis. Drills would also be required to test the adequacy of reactor and plant operating procedures.

This is an effort to reduce the risk of off-normal operating conditions by improving the capability of operators and supervisors to understand and control complex reactor transients and accidents, and also to improve the general capability of an operations organization to respond rapidly and effectively to upset conditions.

PRIORITY DETERMINATION

Assumptions

Assume that the frequency of core-melt incidents is 5×10^{-5}/plant-year, based on WASH-1400.[16] Also, assume that operator error accounts for 50% of these events, but that the plant drills will improve operator performance by 2%. In addition, assume that the release associated with core-melt is the value averaged over the probabilities of the WASH-1400[16] accident categories for PWRs and BWRs and weighted by the number of PWRs (95) and BWRs (48). This results in a total of 2.4×10^6 man-rem per accident. The remaining average plant lifetime is assumed to be 28 years.

Frequency/Consequence Estimate

Based on the assumptions above, the reduction in the core-melt frequency resulting from the plant drills is calculated to be $(0.02)(0.50)(5 \times 10^{-5})$/plant-year or 5×10^{-7}/plant-year.

Risk Reduction = $(5 \times 10^{-7})(2.4 \times 10^6)(28)(143)$ man-rem = 4,805 man-rem

Cost Estimate

Industry Cost: The industry resources required for implementation are estimated to be one person-month per plant. This is the estimated personnel requirement associated with the utility staff time for attendance at the drill, preparation by staff and management, and staff time dedicated to the dissemination of insights gained from the drills. At a cost of $100,000/man-year and with 4.33 weeks per month, this yields a per-plant cost of $8,333. Across the industry, i.e., 143 plants, this would be $1.2M.

The industry resources required annually to participate in the plant drills are estimated to be two person-months per plant, which includes drill attendance, preparation before the drill, and dissemination of information afterward. This would be equivalent to $16,660/plant-year. For the total industry (143 plants), this works out to an estimated 143 person-months/year or $2.38M/year. Given the average remaining lifetime for the plants (28 years), this gives a total operational cost of $67M.

The total cost to industry is then the sum of the implementation and operational costs, $(1.19 + 67)M or approximately $68.2M.

NRC Cost: The total costs to the NRC to implement the resolution of this issue includes NRC staff labor and services of a contractor. Since the activities of the NRC staff and the contractor are to some degree interchangeable, no attempt was made to provide separate estimates so that the total implementation cost is estimated to be $300,000. The annual cost to the NRC was also estimated to be $300,000. Again, this was assumed to contain some mixture of staff and contractor expenses. Over the average remaining life (28 years), the operational cost comes to $8.4M. Therefore, the total cost to the NRC is the sum of implementation and operation costs, $(8.4 + 0.3)M or $8.7M.

Hence, the total costs associated with this issue are $(68.2 + 8.7)M or $76.9M.

Value/Impact Assessment

Based on a public risk reduction of 4,805 man-rem, the value/impact score is given by:

$$S = \frac{4,805 \text{ man-rem}}{\$76.9M}$$

$$= 62 \text{ man-rem/\$M}$$

CONCLUSION

Based on the above value/impact score, the ranking of this issue would be low to medium. However, because the risk may have been estimated to be well on the conservative side, our judgment is that the issue of plant drills should receive a LOW priority ranking.

ITEM I.A.2.6: LONG-TERM UPGRADING OF TRAINING AND QUALIFICATIONS

ITEM I.A.2.6(1): REVISE REGULATORY GUIDE 1.8

Items I.A.2.6(1), I.A.2.6(2), I.A.2.6(3), and I.A.2.6.(5) have been combined and evaluated together.

DESCRIPTION

Historical Background

Item I.A.2.6 of the TMI Action Plan[48] calls for the long-term upgrading of training and qualifications for operations personnel. The specific paragraphs of this item in NUREG-0660[48] call for a revision of "Regulatory Guide 1.8[226] (ANSI/ANS 3.1),"[253] in order to incorporate short-term requirements into this issue and any other changes resulting from a national standards effort. Also, it is stated that more explicit guidance regarding exercises in simulator requalification programs will be included in the regulatory guide (Recommendation 8 of SECY-79-330E[251]) as will qualifications of shift supervisors and senior reactor operators [NUREG-0585,[174] Recommendations 1.6(1) and (2)]. In addition, based on the NRC staff review of NRR-80-117,[252] recommendations will be made to the Commission and Commission decisions will be factored into the regulatory guide or regulation changes. Moreover, appropriate revisions to 10 CFR 55, Operator Licenses, are to be recommended for action by the Commission in order to incorporate the applicable short-term changes plus requirements based on Commission action on SECY-79-330E[251] for mandatory simulator training for applicants for licenses (Recommendation 4); mandatory simulator training in requalification programs (Recommendation 7); NRC administration of requalification examinations (Recommendation 9 as modified by the Commission); and mandatory operating tests at simulators (Recommendation 11). Finally, it is noted that the Nuclear Waste Policy Act of 1982, Public Law 97-425, Section 306 authorized and directed NRC to promulgate regulations or guidance for the training and qualifications of civilian nuclear power plant personnel. A task force has been formed within NRC as a result of this bill. As part of the task force objectives, Items I.A.2.6 (1, 2, and 3) are to be addressed.

The numerical assessment of this safety issue was conducted by the PNL staff[64] with experience in reactor operator licensing, reactor operation, and general reactor safety in consultation with General Physics Corporation. General Physics Corporation provides utility training services and has significant experience in reactor simulators, providing procurement and startup assistance, operation and maintenance services, and simulator modifications.

Safety Significance

A public risk reduction is anticipated as a result of a reduction in core-melt frequency which follows from a reduction in operator error rates. Reduction in operator errors is expected to result from the upgraded training and qualifications which form the assumed resolution of this safety issue.

Possible Solutions

The upgrades are assumed to include an increase in time spent in simulator operation both in training and in requalification. The simulator time is assumed to improve in quality as well as quantity. Emphasis on improvements on the operators' diagnostic capability is felt to be especially important in contributing to a reduction in core-melt frequency. Furthermore, the enforcement activities in term of NRC-administered examinations and OIE inspection of training programs is likely to emphasize the value of this long-term training and qualification of reactor operators.

PRIORITY DETERMINATION

Assumptions

It is assumed that the resolution of this safety issue will take the form of upgrading utility training and qualification programs that will represent a major enhancement of the training and qualification programs.

It is noted that many of the TMI Action Plan Items associated with operator training are interrelated and it is, therefore, difficult to assess them independently. For example, this issue is related to I.A.4.1, Initial Simulator Improvement, which deals with the improvement of simulators and provides for more realistic modeling of the plant whereas this issue, [I.A.2.6(1,2,3,5)], deals with training improvements, including the enhanced use of existing simulators. Either issue, by itself, would improve operator performance. However, there may be significant overlaps in improving operator performance if both items were implemented. Even though it is recognized that the total improvement would be less than the sum of the individual contributions when each is assessed separately, the extent of any overlap is not identified here.

Based on engineering judgment, it was estimated by the PNL panel that the resolution of this safety issue would result in a 30% reduction in operator error rates. The number of plants to which this issue is applicable is assumed to be 95 PWRs and 49 BWRs with average lifetimes of 28.5 years and 27 years respectively.

For the analysis performed by PNL,[64] Oconee-3 is taken as the representative PWR plant. It is assumed that the fractional risk and core-melt frequency reductions for the representative BWR (Grand Gulf) will be equivalent to those for the representative PWR. Therefore, the analysis is conducted only for the PWR but the fractional risk and core-melt frequency reductions are also applied to the BWR. The dose calculations are based on a reactor site population density of 340 people per square mile and a typical mid-west meteorology is assumed.

Frequency/Consequence Estimate

Based on the affected accident sequences and the parameters affected by this safety issue resolution (SIR), the original core-melt frequencies of 8.2×10^{-5} per plant-yr for PWRs and 3.71×10^{-5}/plant-yr for BWRs are calculated to be reduced by about 16%. The associated reduction in public risk is 31 man-rem/plant-yr for PWRs and 37.4 man-rem/plant-yr for BWRs resulting in a total public risk reduction of 132,600 man-rem.

Cost Estimate

Industry Cost: The resolution of this safety issue was assumed to be a major enhancement of the training and qualification programs. The programs would have to be upgraded in order to meet the requirements of INPO accreditation. These requirements are assumed to be far-reaching and require significant effort on the part of utility training staffs. The amount of effort will vary among the utilities, depending on the present state of their programs. The effort required to implement the program is estimated by the PNL panel to require 10 to 20 man-years of effort for each plant. The mean value is expected to be shifted toward the lower end since many utilities are currently improving their training programs. A 12 man-year effort is taken as the central estimate.

Operation under the upgraded programs would require enhanced training activities and more operator time in training. The training staff is estimated to require three additional people. It is assumed the major cost of additional operator time can be estimated from increased time at simulators. It is estimated that 40 hours of simulator time will be added to operator training and requalification. For 20 operators per year passing through these programs, this is equivalent to 800 additional hours. It is further assumed that operators can be trained three at a time on the simulator and that simulator time can be acquired for \$600/hour. This gives an additional simulator cost of \$160,000/year. The industry costs are estimated as follows:

(1) Implementation of the SIR

$$\left(12\ \frac{\text{man-yrs}}{\text{plant}}\right)(49 + 95)\ \text{plants}\ \left(\frac{\$100{,}000}{\text{man-yr}}\right) = \$173\text{M}$$

(2) Operation and Maintenance of the SIR

(a) Labor

$$\text{Training Staff} = \left(\frac{3\ \text{man-yr}}{\text{plant-yr}}\right)\left(52\ \frac{\text{man-wk}}{\text{man-yr}}\right) = 156\ \frac{\text{man-wk}}{\text{plant-yr}}$$

$$\text{Operators} = \frac{(800 \text{ man-hr})}{\text{plant-yr}} / \frac{(40 \text{ man-hr})}{\text{man-wk}} = \frac{20 \text{ man-wk}}{\text{plant-yr}}$$

$$\text{Total Labor} = \frac{176 \text{ man-wk}}{\text{plant-yr}}$$

(b) Simulator Time (Operators)

$$\frac{(800 \text{ man-hr})}{\text{plant-yr}} / \left(\frac{3 \text{ man-hr}}{\text{simulator-hr}}\right) = \frac{267 \text{ simulator-hr}}{\text{plant-yr}}$$

The industry cost per plant-year for operation and maintenance is given by:

$$\left(\frac{176 \text{ man-wk}}{\text{plant-yr}}\right) \times \left(\frac{\$100{,}000/\text{man-yr}}{52 \text{ man-wk/man-yr}}\right) + \left(\frac{267 \text{ simulator-hr}}{\text{plant-yr}}\right) \left(\frac{\$600}{\text{simulator-hr}}\right)$$

= 500,000/plant-year

Therefore, for all affected plants, the total industry cost for operation and maintenance is given by:

($500,000/plant-yr) [(49)(27) + (95)(28.5)] plant-yr = $2,000M

The total industry cost for implementation, operation, and maintenance of the solution is then [$173M + $2,000M] or $2,173M.

NRC Cost: The NRC effort to implement the resolution of this issue would be significant. It is estimated in NUREG-0660[48] that 5.4 man-years plus $259,000 would be required. Some of these development activities have been completed. However, much work remains to be done. The remaining effort is estimated to be 4.5 man-years and $100,000.

The operational activities of the NRC would include reviews of training programs, increase inspection and additional examination. The annual labor for reviews and inspections is estimated to be equivalent to 3 person-years. The principal addition in examinations is assumed to be NRC conduct of a portion of requalification examinations. It is assumed the NRC will conduct 25% of the requalification examinations and the 20 operators are requalified at each plant every year. It is estimated that one person-month is required for each plant. This assumes the five (25% of 20) operators selected for NRC examination at each plant are tested at the same time. NRC costs are estimated as follows:

(1) Implementation of the SIR

Staff Labor + Other Costs
= (1.4 man-wk/plant)($1,600/man-wk) + ($100,000)/144 plants
= $3,386/plant

Total cost for all affected plants is ($3,386/plant)(144 plants) or $488,000.

(2) Review of Maintenance and Operation of SIR

(a) Review and Inspection $= (\frac{3 \text{ man-yr}}{\text{yr}})(\frac{52 \text{ man-wk}}{\text{man-yr}})/144$ plants

$= 1.08$ man-wk/plant-yr

(b) Examination $= (1 \frac{\text{man-month}}{\text{plant-yr}})\ (\frac{3.7 \text{ man-wk}}{\text{man-month}})$

$= 3.7$ man-wk/plant-yr

Total time spent is 4.78 man-wk/plant-yr.

The NRC cost per plant-yr due to review of operation and maintenance is (4.78 man-wk/plant-yr)($1,900/man-wk) = $9,088/plant-yr.

The total NRC cost for operation and maintenance of the SIR is then ($9,088)[(49)(27) + (95)(28.5)] = ($9,088)(4,030) = $36.6M

Therefore, the total industry and NRC costs are estimated to be $[2,173 + 0.488 + 36.6]M or $2,210M

Value/Impact Assessment

Based on the estimated reduction in public risk of 132,600 man-rem, the value/impact score is given by:

$$S = \frac{132,600 \text{ man-rem}}{\$2,210M}$$

$$= 60 \text{ man-rem/\$M}$$

Other Considerations

The total occupational risk reduction is associated only with accident avoidance inasmuch as there is no dose associated with implementation or maintenance of this SIR. With a dose of 20,000 man-rem associated with accident cleanup and with the calculated reductions in core-melt frequencies of 1.3×10^{-5}/plant-yr and 5.9×10^{-5}/plant-yr for PWRs and BWRs, respectively, the total occupational dose reduction is calculated to be 860 man-rem.

CONCLUSION

Although the value/impact score was low, this issue was determined to be high priority because of the large potential public risk reduction. However, with the publication of NUREG-0985, Revision 1,[651] Item I.A.2.6(1) is now covered in Sections 1.2 and 2.1 of the HFPP.

ITEM I.A.2.6(2): STAFF REVIEW OF NRR 80-117

This item was evaluated in Item I.A.2.6(1) above and, in accordance with an RES memorandum,[437] was RESOLVED. No new requirements were established.

ITEM I.A.2.6(3): REVISE 10 CFR 55

This item was evaluated in Item I.A.2.6(1) above and, as a result of the Nuclear Waste Policy Act of 1982 (Public Law 97-425), the scope of this item is now covered under Item I.A.2.2.[438]

ITEM I.A.2.6(4): OPERATOR WORKSHOPS

DESCRIPTION

Historical Background

On the basis of NUREG-0660,[48] NRR is required to develop a Commission paper on training workshops for licensed personnel. NUREG-0585,[174] the source of this safety issue, states that the intent of the issue is to conduct seminar-type workshops to exchange information on operations experience between the NRC and licensees and among licensees. This would assist in the improvement of operator performance and in improvements to reactor regulation, both resulting in improved safety. The proposed requirements would have one representative for each shift at each unit attend such a workshop annually.

Safety Significance

It is expected that there are two potential pathways to improved safety benefit emerging from this issue: (1) improved operator performance through the sharing of safety-related experiences and (2) the effect of improved regulation arising out of interaction between the operators and the NRC attending the workshops. The second pathway is considered to be a second-order effect and very difficult to quantify. Therefore, it was assumed that all the benefit would be derived through the reduction in operator-error rates.

PRIORITY DETERMINATION

Assumptions

PNL has conducted and is conducting a series of these workshops for NRR. In the assessment of this issue, PNL staff responsible for these workshops were consulted. Their judgments form the basis of our analysis.

This analysis assumes the major gains in reactor safety will come through the improvement in operator performance; that is, a reduction in their error rates. There is also a pathway to improve safety by means other than human performance through improved regulations developed from operator input at the workshops. The latter would be extremely difficult to quantify so that only the human error rate-reduction pathway to improved safety will be treated.

A panel of PNL experts was assembled and included staff that conduct operator licensing examinations, staff with experience in reactor operations, reactor safety and risk assessment, and the staff responsible for the conduct of the current operator feedback workshops. This panel produced the estimates that form the basis of this analysis.

The analysis is based on the following additional assumptions:

1. Applicable Plants: 95 PWRs and 48 BWRs

2. Selected Analysis Plant: Oconee 3 - representative PWR. It is assumed that the fractional risk and core-melt frequency reductions for the representative BWR (Grand Gulf) will be equivalent to those for the representative PWR. Therefore, the analysis is conducted only for the PWR, but the fractional risk and core-melt frequency reductions are also applied to the BWR.

3. Affected Accident Sequences and Base-Case Frequencies: Most sequences are affected. The affected sequences and the base-case frequencies are shown in NUREG/CR-2800.[64]

4. Affected Release Categories and Base-Case Frequencies: All release categories are affected by issue resolution. The original base-case frequencies are used as given below.

Oconee	Grand Gulf
PWR-1 = 1.10×10^{-7}/plant-yr	BWR-1 = 1.09×10^{-7}/plant-yr
PWR-2 = 1.0×10^{-5}/plant-yr	BWR-2 = 3.35×10^{-5}/plant-yr
PWR-3 = 2.86×10^{-5}/plant-yr	BWR-3 = 1.44×10^{-6}/plant-yr

Frequency/Consequence Estimate

The PNL panel estimated[64] the most likely reduction in human error rates for operators due to the conduct of the proposed workshops would be 3%. This is assuming the workshops are conducted in the manner now perceived. That is, to focus on data gathering for the NRC. This reduces the amount of time that could be devoted to inter-licensee sharing of operational experiences which would have a more direct effect on safety-related operational performance in the plants. The possible range of reduction stretched from 1% to 10%. If the focus could be shifted toward the inter-licensee exchange of operational experiences, the most likely reduction in error rate would shift upward. However, it is not expected to exceed 10%.

Based on the PNL estimates and calculations,[64] and assuming a typical midwest-type meteorology and an average population density of U.S. reactor sites of 340 people per square mile, the public risk reduction is 7,140 man-rem for 143 plants with an average existing lifespan of 28 years. The occupational dose reduction is minor at a calculated value of 46 man-rem.

Cost Estimate

Industry Cost: The industry resources required for implementation are estimated to be one person-month per plant. This is the estimated personnel requirement associated with the trial workshops currently being conducted. It includes utility staff time for attendance of the workshop, preparation by staff and management, and staff time dedicated to the dissemination of insights gained at the workshop. At a cost of $100,000/person-year and with 4.33 weeks per month, this yields a per-plant cost of $8,333. Across the industry, i.e., 143 plants, this would be $1.19M.

The industry resources required annually to participate in the training workshops are estimated to be the same as those for implementation. That is, one person-month per plant, which includes workshop attendance, preparation before the workshop, and dissemination of information afterward, would be needed. This would be equivalent to $8,333/plant-year. For the total industry (143 plants), this works out to an estimated 143 person-months per year or $1.19M per year. Given the average remaining lifetime for the plants, this gives a total operational cost of $33.3M. Therefore, the total industry cost associated with this issue is $34.5M.

NRC Cost: The total cost to the NRC to implement the resolution of this issue was estimated to be $0.3M. This includes NRC staff labor and services of a contractor. Since the activities of the NRC staff and the contractor are to some degree interchangeable, no attempt was made to provide separate estimates. The annual cost to the NRC was also estimated to be $0.3M. Again, this was assumed to contain some mixture of staff and contractor expenses. Over the average remaining life, the operational cost comes to $8.4M. While not specific, these estimates for implementation and operation are firmly based on the experience of conducting the present trial workshops. Therefore, the total cost to the NRC is the sum of implementation and operation costs which amounts to $8.7M.

Value/Impact Assessment

Based on the estimated public risk reduction of 7,140 man-rem, the value/impact score is given by:

$$S = \frac{7{,}140 \text{ man-rem}}{\$(34.5 + 8.7)\text{M}}$$

$$= 165 \text{ man-rem/\$M}$$

Other Considerations

The accident avoidance cost is the product of the change in accident frequency (ΔF) and the estimated cost to the utility of a major accident (A). This latter term is estimated[64] to be $1.65 Billion. The cost per plant-year is then estimated to be:

PWRs: $(\Delta F)(A) = (7 \times 10^{-7})(\$1{,}650\text{M})/\text{plant-yr} = \$1{,}200/\text{plant-yr}$
BWRs: $(\Delta F)(A) = (3.2 \times 10^{-7})(\$1{,}650\text{M})/\text{plant-yr} = \$530/\text{plant-yr}$

The total cost for all plants is the per-plant-year cost multiplied by the number of plants (N) and the average remaining lifetime (T) for each type of plant:

$$\Sigma(NT)(\Delta F)(A) = \$(95)(28.5)(1{,}200)\text{M} + \$(48)(27.0)(530)\text{M} = \$3.9\text{M}$$

CONCLUSION

Because of the extensive number of sequences considered by PNL to be affected by this issue, the base-case risk is high at a calculated range of from 60 to

73 man-rem/plant-year. With a value/impact score of 165 man-rem/$M and an estimated risk reduction of 7,140 man-rem, this issue should have a MEDIUM priority ranking.

ITEM I.A.2.6(5): DEVELOP INSPECTION PROCEDURES FOR TRAINING PROGRAM

This item was evaluated in Item I.A.2.6(1) above and, in accordance with an OIE memorandum,[379] was RESOLVED. No new requirements were established.

ITEM I.A.2.6(6): NUCLEAR POWER FUNDAMENTALS

DESCRIPTION

This TMI Action Plan item calls for NRR to develop requirements for the inclusion of nuclear power fundamentals within the instruction given to reactor operators. This arose out of a concern[174] that the 12 weeks of fundamentals training given to operators at that time was insufficient.

PRIORITY DETERMINATION

In order to assess this safety issue, a panel of experts was assembled from the PNL staff. This panel was comprised of members experienced in reactor operator licensing, reactor operations, utility field work, and general reactor safety areas. The results of the PNL assessment are contained in NUREG/CR-2800.[64]

Assumptions

The panel felt there had been significant progress across the industry in the area of instruction in nuclear power fundamentals since the issuance of NUREG-0585[174] in 1979. Further increase in emphasis on fundamentals was felt to be unlikely to improve operator performance. The current trend in operator licensing examinations is to stress operational knowledge and de-emphasize fundamentals. This supports the view that further fundamental training would not add to plant safety.

It was assumed that, if implemented, the additional nuclear power fundamentals training would add 4 weeks to the training period. Also, it was assumed that 20 operators complete the training course each year at every plant. In addition, one full-time instructor was assumed to be required. This yields 80-person-weeks for the operators, 44 person-weeks for the instructors, or 124 person-weeks overall per plant each year. To implement this practice an effort equivalent to one year of operation (124 person-weeks) was estimated to be required.

Frequency/Consequence Estimate

Safety issues which deal with operator training can affect the public risk by improvements in the operator safety-related performance. This can lead to a reduction in core-melt frequency and a reduced probabilistic risk. For this safety issue the PNL panel felt that the current level of instruction in nuclear power fundamentals was adequate. Further emphasis of fundamentals was viewed as not likely to improve operator safety performance. Therefore, there would be no measurable public risk reduction associated with the implementation of this issue. The PNL panel also saw no reduction in occupational dose associated with the implementation of the solution.

Cost Estimate

NRC effort to implement the solution is estimated[48] to be 0.4 person-year or approximately 18 person-weeks. No added costs are estimated for operation for the NRC. The review of the additional instruction could be contained in the current routine function thereby causing no added expense.

Value/Impact Assessment

Based on the judgment that there would be no risk reduction resulting from this issue, the value/impact score is zero.

CONCLUSION

In view of the fact that it is believed that the current level of instruction in nuclear power fundamentals is adequate for reactor operators, further emphasis of fundamentals as required by this issue is viewed as not likely to improve operator safety performance. The resulting value/impact score of zero indicates that this issue should be DROPPED from further consideration.

ITEM I.A.2.7: ACCREDITATION OF TRAINING INSTITUTIONS

DESCRIPTION

Historical Background

Based on the requirements of NUREG-0660,[48] this item required NRR to complete a study to establish the procedures and requirements for NRC accreditation of reactor operator training programs. The resulting study would be developed into a Commission paper describing the various options for accreditation.

Safety Significance

There are two aspects to the safety benefit for this issue. One is the reduction of public risk through the improvement of operator performance, which is expected from the improved training accreditation. The second is a reduction in occupational exposure. This will primarily be for operators who often supervise maintenance or perform other duties in radiation zones. However, some reduction in routine occupational exposure can also be expected for other operations personnel as a result of the increased awareness by the operators.

Possible Solution

In order to assess this safety issue, a panel of experts was assembled from the PNL staff. This panel was comprised of members experienced in reactor operator licensing, reactor operations, utility field work, and general reactor safety areas.

The panel envisioned the resolution of this safety issue as the formation of an accreditation board consisting of representatives from the NRC, industry, and academia. This board would develop and apply criteria for accreditation. This would include training programs of utilities, university-related programs, and independent training institutions. While theoretically applying to training for all operations staff, the PNL panel felt the current thrust was focused on reactor operators. Therefore, the assessment was made assuming only operators would be affected.[64]

PRIORITY DETERMINATION

Assumptions

The views of the panel include an awareness of the fact that some training programs are very near to accreditation already. Either through association with the universities or through other means of providing high quality instruction, these programs would be likely to acquire accreditation from the board easily. Other training programs are not so well prepared for accreditation and may require significant effort and expense to upgrade them. Some savings may be gained for multi-unit sites in sharing costs.

Therefore, the resolution of this safety issue was assumed to be an improvement in operator performance. For some utilities, approximately 10% of the total, this issue will have essentially no effect. This is because: (1) their current training programs would be accredited with little effort and (2) the quality of their programs is sufficiently high that accreditation would result in no discernible improvement in their operators' performance. Other utilities will see varying degrees of improvement. Those with training programs that are below the accreditation standards will be brought up nearer to the high quality enjoyed by the outstanding utilities. Overall, the effect on operator human error is estimated to be a reduction of 10% across the affected portion of the industry. The detailed assumptions for this analysis are as follows:

1. Applicable Plants: BWRs and PWRs - 90% of total plants; 43 BWRs, 86 PWRs, or 129 plants in all.

2. Selected Analysis Plant: Oconee 3 - representative PWR. It is assumed that the fractional risk and core-melt frequency reductions for the representative BWR (Grand Gulf) will be equivalent to those for the representative PWR. Therefore, the analysis is conducted only for the PWR, but the fractional risk and core-melt frequency reductions are also applied to the BWR.

Frequency/Consequence Estimate

Based on the PNL analysis,[64] and assuming a typical midwest-type meteorology and an average population density of U.S. reactor sites of 340 people per square mile, the anticipated public risk reduction is calculated to be 26,180 man-rem.

Cost Estimate

The PNL panel estimated[64] the costs associated with implementation and operation of the resolution to this safety issue. The one-time costs to industry to implement the change initially was estimated to be in the range of \$0.1M to \$1M per reactor. Those with training programs closer to accreditable status would enjoy the smaller costs. The best estimate for the average plant was taken to be \$0.3M. Operation under the accreditation program was estimated to cost between \$0.05M and \$0.25M per plant annually for additional funding to maintain an accredited training program. The best estimate was \$0.1M per plant annually.

The cost to the NRC to implement the accreditation was estimated to be \$0.635M which is equivalent to 330 person-weeks. The annual operational cost to the NRC is estimated[64] to be \$100,000 or one person-year.

The detailed breakdown of these costs are as follows:

<u>\$300,000/Plant Industry Implementation (approximately 3 man-yrs):</u>

- to review accreditation standards
- to compare the present utility practices with the developed standards
- plan the necessary upgrades
- implement the program upgrades to fulfill the accreditation requirements.

<u>\$100,000/Plant-yr Industry Operation and Maintenance:</u>

- time invested by the staff in upgraded training (increased course time, quality, etc.)
- instruction upgrade (time, quality, etc.)

<u>\$500,000 NRC Implementation (approximately 5 man-yrs)</u>

- predicated on the possibility that INPO accreditation will not be forthcoming; NRC may have to do
- NRC to develop accreditation standards, regulations, and implement to adoption by the industry.

<u>\$100,000 NRC Operation and Maintenance (approximately 1 man-yr/yr)</u>

- additional OIE efforts to assure industry maintenance of standards (all plants).

The total costs for this safety issue are, therefore, estimated[64] by PNL as follows:

1.	Implementation of the Safety Issue Resolution (SIR) by industry	\$ 39,000,000
2.	Operation and Maintenance of the SIR by the industry	360,000,000
3.	NRC Implementation of the SIR	635,000
4.	NRC Operation and Maintenance of SIR	2,800,000
	Total:	\$402,435,000

Value/Impact Assessment

Based on the estimated public risk reduction of 26,180 man-rem, the value/impact score is given by:

$$S = \frac{26{,}180 \text{ man-rem}}{\$402.4M}$$

$$= 65 \text{ man-rem/\$M}$$

Other Considerations

The industry accident avoidance cost was estimated by PNL[64] to be $14M. The occupational risk reduction is estimated to be 22,170 man-rem resulting from accident avoidance (170 man-rem) and from operation and maintenance of the safety issue resolution (22,000 man-rem).

CONCLUSION

Although the value/impact score was low, this issue was determined to be medium priority because of the magnitude of the potential public risk reduction. However, with the publication of NUREG-0985, Revision 1,[651] the issue is now covered in Section 2.2 of the HFPP.

REFERENCES

16. WASH-1400 (NUREG-75/014), "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, October 1975.

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980.

54. NUREG/CR-1659, "Reactor Safety Study Methodology Application Program," U.S. Nuclear Regulatory Commission, 1981.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

174. NUREG-0585, "TMI Lessons Learned Task Force Final Report," U.S. Nuclear Regulatory Commission, October 1979.

226. Regulatory Guide 1.8, "Personnel Selection and Training," U.S. Nuclear Regulatory Commission, May 1977.

251. SECY-79-330E, "Qualifications of Reactor Operators," July 30, 1979.

252. NRR-80-117, "Study of Requirements for Operator Licensing," February 4, 1980.

253. ANSI/ANS 3.1, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants," American National Standards Institute, 1981.

379. Memorandum for H. Denton from R. DeYoung, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," January 24, 1983.

437. Memorandum for H. Denton from R. Minogue "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," March 29, 1983.

438. Memorandum to Office Directors from W. Dircks, "NRC Actions Required by Enactment of the Nuclear Waste Policy Act of 1982," January 19, 1983.

651. NUREG-0985, Revision 1, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," U.S. Nuclear Regulatory Commission, September 1984.

TASK I.A.3: LICENSING AND REQUALIFICATION OF OPERATING PERSONNEL

The objectives of this task are as follows: (1) to upgrade the requirements and procedures for nuclear power plants operator and supervisor licensing to assure that safe and competent operators and senior operators are in charge of the day-to-day operation of nuclear power plants, and (2) to increase the requirements for initial issuance of licenses and for license renewals and provide closer NRC monitoring of licensed activities.

ITEM I.A.3.1: REVISE SCOPE OF CRITERIA FOR LICENSING EXAMINATIONS

DESCRIPTION

This NUREG-0660[48] item called for NRR to notify all operator license holders and applicants of the new scope of examinations and criteria for issuance of reactor operator (RO) and senior reactor operator (SRO) licenses and renewal of licenses. Simulator examinations were to be included as part of the license examination. Clarifications to this item were issued in NUREG-0737.[98]

CONCLUSION

This item was resolved and requirements were issued. However, as a result of P.L. 97-425, it was determined that additional staff work on the issue was required and a proposed rule for operator licensing was presented to the Commission in SECY-84-76.[593] Approval of this rule would effectively close out this item.

ITEM I.A.3.2: OPERATOR LICENSING PROGRAM CHANGES

DESCRIPTION

This TMI Action Plan item[48] called for NRR to take the following actions:

(1) Develop and implement a plan to relocate Operator Licensing Branch (OLB) examiners at Nuclear Power Plant Simulator Training Centers or in Inspection and Enforcement Regions.
(2) Conduct a study of the staffing of the operator licensing program and the qualifications and training of examiners.
(3) Develop and implement a plan to report operator errors and to act on operator errors with respect to continuation of licensing.

As a result of the above actions, the following accomplishments were made:

(1) "The administering of examinations and issuance/renewal of operator licensing will be transferred to Region III in FY 1982 and to Region II in FY 1983. All regions will have operator licensing authority in FY 1984. NRR will provide oversight and guidance, including examination procedures and criteria."[88]

(2) A study of the staffing of the operator licensing program and the qualifications and training of examiners was completed in November 1980 and documented in NUREG/CR-1750.[89]
(3) A plan for reporting operator errors and for acting on operator errors with respect to continuation of licensing was developed in NUREG/CR-1750.[89] However, after review of this recommended plan, DHFS concluded that no further action was required.[440]

CONCLUSION

This item has been RESOLVED and no new requirements were established.

ITEM I.A.3.3: REQUIREMENTS FOR OPERATOR FITNESS

DESCRIPTION

Historical Background

This safety issue as described in NUREG-0660[48] calls for the NRC to develop a regulatory approach to: (1) provide assurance that applicants for RO and SRO licenses are psychologically fit, and (2) prohibit licensing of persons with histories of drug and alcohol abuse or criminal backgrounds. The regulations will be applied to all current and future operating power plants.

The accomplishments in the program include the publication of NUREG/CR-2075[289] and NUREG/CR-2076.[290] Additionally, a proposed rule addressing alcohol and drug use and the broader issue of fitness for duty of operating licensee personnel and contractors was concurred in by several NRC offices and forwarded to the EDO on April 16, 1982. The proposed fitness for duty rule was issued for public comment in the Federal Register on August 15, 1982, with the public comment period extending to October 5, 1982. A final rule package was completed on December 1, 1982 and a final rule was expected to be published by April 1, 1983. The rule, if promulgated, would require facilities licensed under 10 CFR Part 50.21(b) or Part 50.22 to establish and implement adequate written procedures to provide reasonable assurance that persons with unescorted access to protected areas of nuclear power plants, while in those areas, are not under the influence of alcohol, other drugs or otherwise unfit for duty due to mental or physical impairments. Secondly, a proposed rule amending 10 CFR Part 73.56 regarding access authorization for nuclear power plants has not been completed, although a value/impact analysis in support of the proposed rule has been prepared by the NRC staff.

This issue was assessed by PNL[64] in consultation with a number of engineers who have expertise in reactor operator licensing, reactor operations, utility field work, and general reactor safety areas.

Safety Significance

There could be significant damage if impaired personnel were performing critical safety operations. Legal and institutional problems may limit a thorough implementation of the proposed program. Given that an adequate program were implemented at all power plants and integrated into overall plant operations, the new program would reduce operator error which in turn would lower the risk associated with operation of the power plant.

Possible Solutions

This issue has two components: the first involves initial access to protected areas of nuclear power plants and the second involves continuing fitness for duty once initial access has been granted. The proposed fitness for duty rule, issued for public comment on August 15, 1982, is directed toward the second component of this issue, mandating behavioral observation programs for power plants licensed by the NRC. Behavioral observation is also a part of the proposed Access Authorization Rule directed toward the first component of this safety issue.

The second component of this safety issue deals with limiting access of psychologically unstable individuals to vital plant areas. This component will have a major cost impact on the industry because this access authorization program is comprehensive in that it is aimed at limiting the access to vital plant areas of disgruntled employees, psychologically unsuitable employees, as well as personnel under the influence of drugs or alcohol.

The access authorization program has the following three parts: (1) background search, (2) psychological assessment, and (3) behavior observation. The first two parts would occur prior to granting an individual an unescorted access authorization to protected and vital areas, and the last part would be an ongoing activity for individuals who have been granted an unescorted access authorization. The background check would examine an individual's past for unstable activities, a criminal record, credit problems, and previous employment problems. It has been established by NRC personnel that data on psychological screening shows that for white-collar workers, 2 to 3% are identified as unstable and that for blue-collar employees, the rate is 7 to 10%. These figures provide a background for the assumptions to be made in the priority determination.

PRIORITY DETERMINATION

Assumptions

The major result of this safety issue was assumed to be a reduction in operator error. For some utilities, this new system may result in some reduction in operator error whereas in others the system it may have no discernible effect. Based on engineering judgment, an average of about 2% was arrived at by PNL to apply to all currently operating and future plants. Thus, this issue assumes the implementation of the access authorization system at all 134 plants either under construction (63) or already in operation (71), with average lifetimes of 28.8 yrs for 90 PWRs and 27.4 yrs for 44 BWRs. Thus, the total remaining life of the affected plants is [(28.8)(90) + (27.4)(44)]RY or 3,798 RY.

Neither the implementation, operation, or maintenance of this SIR would involve any changes in occupational dose accrued by any personnel.

For the analysis performed by PNL,[64] Oconee 3 is taken as the representative PWR. It is assumed that the fractional risk and core-melt frequency reductions for the representative BWR (Grand Gulf) will be equivalent to those for the representative PWR. Therefore, the analysis is conducted only for the PWR, but the fractional risk and core-melt frequency reductions are also applied to the BWR.

Frequency/Consequence Estimate

All release categories are affected by this safety issue but the principal release categories affected by the SIR are 3, 5, and 7. The numerical calculations are based on these categories. The dose calculations are based on a reactor site population density of 340 people per square mile and a typical midwest meteorology is assumed.

The calculated reduction in core-melt frequencies are 4×10^{-7}/RY for PWRs and 1.8×10^{-7}/RY for BWRs. Based on this, the total estimated public risk reduction is 16,000 man-rem. The occupational risk reduction for implementation, operation, and maintenance is zero.

Cost Estimate

Industry Cost: A value/impact analysis in support of the anticipated rule of access authorization has been prepared by the NRC staff and cost estimates for industry have been developed. These cost estimates, which have been reviewed and accepted by AIF, are as follows:

(1) For all existing plants, the implementation cost is $140,000/plant and includes the preparation of the plant and associated procedures ($33,000), licensee management and clerical staff ($63,000), training to implement the behavioral observation program ($34,000), and storage for files ($10,000). The total industry implementation cost for existing plants is $(140,000)(71) = $9.94M.

(2) For all future plants (in which none of the employees will be grandfathered), the implementation costs are estimated to be $590,000 per plant. In addition to the costs noted above for existing plants, this implementation includes the cost of background investigations ($375,000), review process and appeals procedures ($36,000), increased file storage requirements ($30,000), and miscellaneous criminal checks with the FBI, etc. ($9,000). The total industry cost for future plants is ($590,000)(63) = $37.2M.

(3) The cost of operation of the access authorization system at each plant is estimated to be $300,000/year. This operating cost includes background investigations for new people as a result of employee turnover ($94,000), professional management and clerical staff ($63,000), review and appeal process ($67,000), refresher training for old supervisors ($19,000), training of new supervisors ($9,000), plan maintenance and updates ($8,000), file storage ($39,000), and criminal history checks with the FBI for new people ($2,000). The total industry cost for operation and maintenance of the access authorization system is ($0.3M/RY)(3,798 RY) or $1,140M.

The total industry cost for the SIR is $[1,140 + 9.94 + 37.2]M or $1,187M.

NRC Cost: The NRC costs for the SIR are estimated as follows:

(1) The NRC time for further development and issuance of the proposed plan is estimated to be 1.5 man-years. At a rate of $100,000/man-year, the estimated cost for this effort is $150,000.

(2) For implementation of the plan, which includes the review and modification of the utilities' plans, the NRC effort was estimated to be 1.5 man-years. For the 134 affected plants, this amounts to 0.6 man-week/plant. At a cost of $2,270/man-week, the NRC implementation cost is $182,500.

(3) NRC review of the operation and maintenance of the SIR is estimated to require 1 man-week/RY for all plants. At a cost of $2,270/man-week, the total NRC cost for operation and maintenance of the SIR is $8.6M.

The total NRC cost for the SIR is $[0.15 + 0.1825 + 8.6]M = $8.9M.

Value/Impact Assessment

Based on a public risk reduction of 16,000 man-rem, the value/impact score is given by:

$$S = \frac{16{,}000 \text{ man-rem}}{\$(1{,}187 + 8.9)\text{M}}$$

$$= 13.4 \text{ man-rem/\$M}$$

Other Considerations

It has been estimated by cognizant personnel at the NRC that the Fitness for Duty Rule will have a negative cost impact on operating licensees in the long run. The NRC estimates that initial licensee burden to develop written procedures required by the rule will be approximately 1,200 man-hours over a six-month period at a total cost between $50,000 and $75,000, if no fitness for duty program exists at the licensee's facility. While utilities such as TVA claim that alcohol abuse alone costs them approximately $18.5M annually, fitness for duty programs of the type envisioned by the Fitness for Duty Rule are expected to save costs through quicker identification of employees not fit for duty and through assisting these employees, in whom considerable resources have been invested, so that they might return to high levels of productivity. Absenteeism due to alcohol-drug abuse costs U.S. industry an average of $300 annually for every worker nationwide. Alcohol drug-abusers lose an additional 25% of their productive time when on the job, at an average annual cost to U.S. industry of approximately $2,900 per abuser. The total annual cost to U.S. industry is between $12 billion to $15 billion. Wrich, in "The Employee Assistance Program; Updated for the 1980's," Hazelden, 1980, reports that U.S. industry receives a return of $10 in decreased absenteeism, accidents, and increased productivity for every dollar it spends on fitness for duty.

CONCLUSION

Although the estimated risk reduction was 16,000 man-rem and the value/impact score only 13.4 man-rem/$M, this issue was given a high priority because of its advanced state of completion. However, with the publication of NUREG-0985, Revision 1, [651] this item is now covered in Section 1.4 of the HFPP.

ITEM I.A.3.4: LICENSING OF ADDITIONAL OPERATIONS PERSONNEL

DESCRIPTION

Historical Background

This TMI Action Plan item[48] seeks to upgrade the operations performance in nuclear power plants by imposing licensing requirements upon other operations personnel in addition to ROs and SROs.

Safety Significance

It is possible that, by undergoing licensing, personnel such as managers, engineers, and technicians would be better qualified and less likely to commit errors in performing their functions.

Possible Solution

A study could be undertaken to determine which, if any, personnel should be licensed. Licensing would then be required by the NRC for those additional personnel.

PRIORITY DETERMINATION

Assumptions

It was estimated that the effects of resolution of this issue would be minimal for many utilities since there are existing practices which go a long way toward ensuring that qualified and trained individuals are in the responsible positions. It was assumed that additional licensing requirements would produce some improvement by assisting in the screening of potentially poor performers from the operations staff. The net effect was estimated to be equivalent to a 2% reduction in human error rates for reactor operators and maintenance personnel.[64]

Frequency Estimate

Based on the 2% reduction in human error rate, the Oconee 3 (representative PWR) risk equation parameters were adjusted. All Accident Sequences except V were assumed to be affected and all Release Categories were affected. The reduction in core-melt frequency for Oconee 3 was calculated to be 1.4×10^{-6}/RY. The reduction in core-melt frequency for Grand Gulf 1 was then calculated by assuming that the fractional core-melt frequency reduction for the representative BWR will be equivalent to the fractional reduction for the PWR. Therefore, since the Oconee 3 fractional reduction was 0.017, the core-melt frequency reduction for Grand Gulf 1 was calculated to be 6.3×10^{-7}/RY.

Consequence Estimate

The corresponding reduction in public risk for Oconee 3 was calculated to be 2.4 man-rem/RY and the public risk reduction for Grand Gulf 1 was calculated to be 2.7 man-rem/RY.

The risk reduction for each type of plant is given as follows:

PWRs: (28.5 yrs)(95 reactors)(2.4 man-rem/RY) = 6.5×10^3 man-rem
BWRs: (27 yrs)(49 reactors)(2.7 man-rem/RY) = 3.6×10^3 man-rem

Therefore, the total risk reduction for this issue is 1.01×10^4 man-rem.

Cost Estimate

Industry Cost: It was assumed that the required additional effort to license the majority of the operations personnel at a plant would be roughly equivalent to the current licensing efforts for ROs and SROs. This was estimated to be $250,000/plant. For operation, industry would have to provide new training staff, staff time for training and exams, and administration. This was estimated to be $50,000/plant-yr. Therefore, the total industry cost is $250M.

NRC Cost: To implement this requirement, the NRC would have to prepare qualification criteria, licensing exams, and procedures. This would be a major undertaking. The NRC costs for implementation were estimated to be in the range of $20M to $50M. For analysis purposes, $35M was used. To operate with the new licensing requirements, it was estimated that the NRC would need 50 additional staff members at a total cost of $5M/year. To perform the annual operational needs of the program, funds would be needed for travel, publications, etc. This was estimated to be an additional $2M/year. Therefore, the total NRC cost is approximately $240M.

Value/Impact Assessment

Based on a total public risk reduction of 10,100 man-rem, the value/impact score is given by:

$$S = \frac{10{,}100 \text{ man-rem}}{\$(240 + 250)\text{M}}$$

$$= 20 \text{ man-rem/\$M}$$

Uncertainty

Because the estimate of the value/impact score relies heavily on the estimated value of the possible reduction in human error rate, the effective improvement may vary significantly.

Other Considerations

DHFS has been pursuing this issue and the Commission has concluded[181] that licensing of managers should not be required. The other portion of the issue (i.e., licensing of other personnel--engineers, maintenance personnel, etc.) is still under study and is to be concluded in FY 1983.

CONCLUSION

Although the value/impact score was low, the potential for risk reduction was considered and this issue was given a medium priority. However, with the publication of NUREG-0985, Revision 1,[651] this item is now covered in Section 1.2 of the HFPP.

ITEM I.A.3.5: ESTABLISH STATEMENT OF UNDERSTANDING WITH INPO

DESCRIPTION

As a part of the overall evaluation of the TMI incident, it was determined[48] that a statement of understanding was needed to address the mutual intent of NRC and INPO concerning the extent to which NRC should review or rely upon training, certification, and other activities of INPO. Consideration was also to be given to providing alternative mechanisms for industry to inform NRC of its general progress on needed safety reforms. It was intended that the statement of understanding would provide a basis for evaluation of any safety reforms or programs. There is no direct risk that can be attributed to this issue.

CONCLUSION

A Memorandum of Agreement[148] between INPO and NRC was issued in April, 1982. However, it did not specifically address training and certification. Following this, the EDO agreed with a revision[594] of Appendix Four to the Memorandum of Agreement (Coordination Plan for NRC/INPO Training-Related Activities) in November 1983. As a result, this Licensing Issue has been resolved.

REFERENCES

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

88. Memorandum for All Employees from H. Denton, "Regionalization of Selected NRR Functions," June 15, 1982.

89. NUREG/CR-1750, "Analysis, Conclusions, and Recommendations Concerning Operator Licensing," U.S. Nuclear Regulatory Commission," January 1981.

651. NUREG-0985, Revision 1, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," U.S. Nuclear Regulatory Commission, September 1984.

148. "Memorandum of Agreement Between the Institute of Nuclear Power Operations and the U.S. Nuclear Regulatory Commission," Rev. 1, April 1, 1982.

181. SECY-82-155, "Public Law 96-295, Section 307(B), Study of the Feasibility and Value of Licensing Nuclear Plant Managers and Senior Licensee Officers," April 12, 1982.

289. NUREG/CR-2075, "Standards for Psychological Assessment of Nuclear Facility Personnel," U.S. Nuclear Regulatory Commission, July 1981

290. NUREG/CR-2076, "Behavioral Reliability Program for the Nuclear Industry," U.S. Nuclear Regulatory Commission, July 1981.

440. Memorandum for W. Minners from D. Ziemann, "Schedules for Resolving and Completing Generic Issues," April 5, 1983.

593. SECY-84-76, "Proposed Rulemaking for Operator Licensing and for Training and Qualifications of Civilian Nuclear Power Plant Personnel," February 13, 1984.

594. Letter to E. Wilkinson (INPO) from W. Dircks (NRC), November 23, 1983.

651. NUREG-0985, Revision 1, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," U.S. Nuclear Regulatory Commission, September 1984.

TASK I.A.4: SIMULATOR USE AND DEVELOPMENT

The objectives of this task are as follows: (1) to establish and sustain a high level of realism in the training and retraining of operators, including dealing with complex transients involving multiple permutations and combinations of failures and errors, and (2) to improve operators' diagnostic capability and general knowledge of nuclear power plant systems.

ITEM I.A.4.1: INITIAL SIMULATOR IMPROVEMENT

ITEM I.A.4.1(1): SHORT-TERM STUDY OF TRAINING SIMULATORS

DESCRIPTION

The TMI Action Plan[48] called for a short-term study of training simulators. The purpose was to collect and develop corrections for presently identified weaknesses. A study of training simulators was undertaken and a report, NUREG/CR-1482,[299] was issued in June 1980.

CONCLUSION

This item has been RESOLVED and no new requirements were established.

ITEM I.A.4.1(2): INTERIM CHANGES IN TRAINING SIMULATORS

DESCRIPTION

The TMI Action Plan[48] stated that requirements to correct specific training simulator weaknesses should be developed based on the short-term study resulting from Item I.A.4.1(1). This item was completed with the issuance of Regulatory Guide 1.149,[439] "Nuclear Power Plant Simulators for Use in Operator Training."

CONCLUSION

This item has been RESOLVED and new requirements were established.

ITEM I.A.4.2: LONG-TERM TRAINING SIMULATOR UPGRADE

The four parts of this item have been combined and evaluated together.

DESCRIPTION

Historical Background

Nuclear power plant simulators are recognized as an important part of reactor operator training. The TMI Action Plan[48] called for a number of actions to improve simulators and their use. There is significant interaction among the simulator-related action items and clear separation is difficult.

Item I.A.4.2 has a number of components dealing with long-term upgrades. The NUREG-0660[48] description calls for research to improve the use of simulators in training operators, develop guidance on the need for and nature of operator action during accidents, and gather data on operator performance. Specific research items mentioned include simulator capabilities, safety-related operator action, and simulator experiments. The item also calls for the upgrading of training simulator standards, specifically updating of ANSI/ANS 3.5-1979. A regulatory guide endorsing that standard and giving the criteria for acceptability is also mentioned. The final portion of Item I.A.4.2 calls for a review of simulators to assure their conformance to the criteria.

A significant portion of the activities to be conducted under this action plan item has been completed. For example, ANSI/ANS 3.5 was revised and issued in 1981. The regulatory guide endorsing this standard, Regulatory Guide 1.149,[439] "Nuclear Power Plant Simulators for Use in Operator Training," as well as numerous research reports have been published.

It is clear that the regulations, the ANS standard, and the regulatory guide do not require a site-specific simulator. 10 CFR 55 states that, if a simulator is used in training, it "... shall accurately reproduce the operating characteristics of the facility involved and the arrangement of the instrumentation and controls of the simulator shall closely parallel that of the facility involved." ANSI/ANS 3.5-1981 calls for a high degree of fidelity between the simulator and the "reference plant." However, there is no requirement that the reference plant be the same facility that the personnel in training will in fact operate. Regulatory Guide 1.149[439] explicitly makes the distinction stating "... the similarity that must exist between a simulator and the facility that the operators are being trained to operate is not addressed in the guide and should not be confused with the guidance provided that specifies the similarity that should exist between a simulator and its reference plant."

The work that has been completed for Item I.A.4.2(1) includes the issuance of NUREG/CR-2353[300] (Volumes I and II), NUREG/CR-1908,[416] NUREG/CR-2598,[417] NUREG/CR-2534,[418] NUREG/CR-3092,[419] and NUREG/CR-3123.[653] This item, however, has long-range requirements calling for: (1) the review of operating experience to provide data on operator responses, and (2) the design and conduct of experiments to determine operator error rates under controlled conditions. Therefore, this item is not completed at this time. However, Items I.A.4.2(2) and I.A.4.2(3) have been completed with the issuance of Regulatory Guide 1.149.[439] Item I.A.4.2(4) concerns the long-term training simulator improvement criteria which were also established in Regulatory Guide 1.149,[439] issued in April 1981, and the criteria were initiated in FY 1982. However, the review of submittals from simulator owners for conformance with the criteria is an on-going task which is still not complete. Therefore, the outstanding portions of this issue that have yet to be completed are the continuation of simulator research and the review for conformance to acceptability criteria.

The assessment of this safety issue was conducted by PNL staff[64] with experience in reactor operator licensing, reactor operation, and general reactor safety, in consultation with General Physics Corporation. General Physics Corporation provides utility training services and is greatly experienced in reactor simulators, providing procurement and startup assistance, operation and maintenance services, and simulator modifications.

In the assessment of this issue it is necessary to acknowledge that many of the TMI action items associated with operator training are interrelated and that ranking problems become involved when an attempt is made to assess these independently. For example, the present issue relates to Items I.A.2.6(1,2,3, and 5), which deal with training improvements including the enhanced use of existing simulators, and I.A.4.1, which deals with initial simulator improvement, including short-term and interim changes in training simulators. However, it is useful to note that the final safety ranking of this issue is relatively insensitive to changes in the basic assumptions used to distinguish these inter-related issues, by the very nature of the ranking matrix. Therefore, it is possible to establish a priority ranking for this issue, despite the possible overlapping of potential benefits and costs with the other inter-related issues.

Safety Significance

Use of simulators with high fidelity to the reference plant would significantly improve operator training in dealing with abnormal conditions thereby reducing operator error. The operators' performance under accident conditions is expected to be enhanced. Thus, potential core melts would be avoided and overall core-melt frequency reduced.

Possible Solution

A possible solution would be to establish a high level of realism in the training and retraining of plant operators by developing simulators with a high degree of fidelity to the reference plant.

PRIORITY DETERMINATION

Assumptions

It was assumed that the major effect of these issues, both in terms of safety benefit and cost incurred, would be in the enhancement of the level of realism imparted by simulators. The specific modeling capabilities given under Item I.A.4.1(2) and in the specification of ANSI/ANS 3.5-1981 specify this feature.

It was assumed for the resolution to this safety issue, that in order to provide the intended level of realism, site-specific simulators would be acquired. Such simulators would be significantly more realistic when compared to the specific facilities, both in layout and operation, than existing generic simulators. In addition, they are assumed to have enchanced transient and accident modeling capabilities.

In our assessment, it was clear that provision of site-specific simulators, while not explicitly required, would meet the requirements of Item I.A.4.1(2), the fidelity requirements of ANSI/ANS 3.5-1981, and the accurate reproduction requirements of 10 CFR 55. Less sweeping simulator enhancements might also fulfill these requirements but would have to be decided on a case-by-case basis. Therefore, for risk, dose, and cost estimates we assumed the enhancement would be effected by the introduction of site-specific simulators.

The public risk reduction (and occupational dose reduction due to accident avoidance) are associated with the reduction in operator error expected to result from the training and requalification of operators on improved simulators. Inasmuch as any studies relating human error rates to the realism of simulator training are not available, this assessment will be based primarily on PNL engineering judgment. Therefore, it is estimated that a reduction in operator error rates of 30% will result from the resolution of this safety issue. This sole-value estimate implies that for specific instances the improvement could be much greater but the 30% reduction is used as an estimate of the average improvement for the purposes of calculation.

The number of plants and the average remaining lifetimes are taken as 90 plants and 28.8 yrs for PWRs and 44 plants and 27.4 years for BWRs. The plants selected for analysis are the Oconee 3 as representative of the PWRs and Grand Gulf as representative of the BWRs. (It is assumed that the fractional risk and core-melt frequency reductions for Grand Gulf will be equivalent to those for the PWR which is calculated directly.)

The dose calculations are based on a reactor site population density of 340 people per square mile and a typical midwest meteorology is assumed.

Frequency Estimate

All release categories are affected by the resolution of this issue. The calculated core-melt frequencies are 8.2×10^{-5}/plant-yr for PWRs and 3.7×10^{-5}/plant-yr for BWRs. The reduction in these frequencies, based on the 30% reduction estimated for operator error, is 1.3×10^{-5}/plant-yr for PWRs and 5.9×10^{-6}/plant-yr for BWRs.

Consequence Estimate

The resulting total reduction in public risk is 150,000 man-rem. The estimated reduction in occupational dose is 820 man-rem based on accident avoidance only since there are no implementation or maintenance dose reductions associated with resolution of this issue.

Cost Estimate

Industry Cost: The major effect of the resolution of these safety issues was assumed to be the acquisition and use of site-specific simulators. The costs to industry of such an undertaking would be substantial. It is important to recognize that if improved modelling changes were possible on existing simulators, the cost to industry would be substantially smaller. However, this is not clear at this time and it is assumed that new simulators would be required. (The impact of this assumption can be weighed subsequently in the final safety priority ranking. The assumption can be reevaluated at that time for any appropriate modifications.)

Assuming that new simulators would be required, the principal industry costs for implementation of this safety issue would be the purchase of the simulators and provision of the new training materials. The capital cost of a simulator is estimated to be $7M. The provision of training materials is estimated to be equivalent to a 7 man-year effort.

It was assumed that all reactors, both operating and planned, would be affected. However, not every reactor would require a simulator. Many reactor sites have two or more reactors located together. If these reactors are sufficiently similar, a single simulator could serve them. Examining the list of 134 operating and planned power reactors, it was estimated that 62 additional site-specific simulators would be adequate. This assumed that 20% of the potential simulators are not required because either a site-specific simulator already exists or the plant in question is an older facility with limited lifetime remaining.

The costs for the 62 new simulators spread over 134 reactors yields $3.2M/reactor in capital cost and 3.2 man-year/reactor to provide new training materials. The operation and maintenance of the new simulators is estimated to require 3 man-years of effort per simulator. Again, sharing the expense for 62 simulators over 134 reactors yields 1.4 man-years/reactor. Industry may also experience costs stemming from participation in simulator experiments and research. However, in comparison to the costs related to new simulators, these costs would be small.

Based on these assumptions the total industry costs are obtained as follows:

(1) Safety Issue Resolution (SIR) Implementation

(a) Labor: $\left(\frac{7\text{ man-yr}}{\text{simulator}}\right)\left(\frac{62\text{ simulators}}{134\text{ plants}}\right)\left(\frac{\$100{,}000}{\text{man-year}}\right) = \$320{,}000$ per plant

(b) Equipment: $\left(\frac{62\text{ simulators}}{134\text{ plants}}\right)\left(\frac{\$7\text{M}}{\text{simulator}}\right) = \$3.2\text{M}$ per plant

Thus, the total industry cost for implementation is (134 plants) ($320,000/plant + $3,200,000/plant) or $470M.

(2) Operation and Maintenance of the SIR

$\left(1.4\ \frac{\text{man-yr}}{\text{reactor}}\right)\left(\frac{\$100{,}000}{\text{man-yr}}\right)$ [(90 PWRs)(28.8 yrs) + (44 BWRs)(27.4 yrs)]

= $530M

Therefore, the total combined industry cost is $(470 + 530)M or $1,000M.

NRC Cost: The principal costs to the NRC are the continuation of research and the conduct of the confirmatory reviews. No additional development costs are foreseen as ANSI/ANS 3.5 is currently being revised and will necessitate a revision to Regulatory Guide 1.149.[439]

The continuing research is treated as an implementation cost. It is estimated to require one NRC man-year and $1M in contractor support. (This includes the remaining costs associated with Item I.E.8.) The confirmatory reviews are also treated as an implementation cost and are estimated to require 4 man-weeks/simulator, or 248 man-weeks in all for the assumed 62 new simulators.

The operational review cost to the NRC is minimal. It is assumed that annually each simulator will be audited to assure that reference plant updates have been adequately represented on the simulator. Such an annual review is estimated to require 2 man-weeks/simulator or 124 man-weeks/year for all 62 new simulators assumed.

NRC costs are estimated as follows:

(1) SIR Development

There is no cost for SIR development since all work is essentially complete and a solution has been identified.

(2) SIR Implementation

(a) Continuing Research: $\frac{\text{1 man-yr}}{\text{134 plants}} = 0.33 \frac{\text{man-wk}}{\text{plant}}$

(b) Initial Simulator Reviews: $\frac{\text{248 man-wk}}{\text{134 plants}} = 1.9 \frac{\text{man-wk}}{\text{plant}}$

Based on a total NRC manpower of 2.23 man-wk/plant, the NRC manpower cost for implementation is

$$\left(\frac{\text{2.23 man-wk}}{\text{plant}}\right) \left(\frac{\$2,270}{\text{man-wk}}\right) \text{(134 plants)} = \$678,300$$

(c) NRC Contractor Support = $1M

Therefore, total NRC Cost for SIR Implementation is ($678,300 + $1M) or $1.7M.

(3) Review of SIR Operation and Maintenance

$$\left(\frac{\text{2 man-wk}}{\text{simulator-yr}}\right) \left(\frac{\text{67 simulators}}{\text{134 plants}}\right) \left(\frac{\$2,270}{\text{man-wk}}\right) = \$2,100\text{/plant-yr.}$$

The total NRC cost for review of SIR operation and maintenance for all affected plants is [(90 PWRs)(28.8 yr) + (44 BWRs)(27.4 yrs)]($2,100/plant-yr) = $8M. Thus, the total NRC cost is $(1.7 + 8)M or $9.7M.

Therefore, total industry and NRC cost for the SIR is $(1,000 + 9.7)M or $1,010M.

Value/Impact Assessment

For a public risk reduction of 150,000 man-rem, the value/impact score is given by:

$$S = \frac{\text{150,000 man-rem}}{\$1,010\text{M}}$$

$$= 148.7 \text{ man-rem/\$M}$$

CONCLUSION

Based on the estimated risk reduction of 150,000 man-rem and the value/impact score of approximately 150 man-rem/$M, the safety priority ranking of this issue would be HIGH. In view of the large estimated risk reduction, this safety priority ranking is essentially unaffected by any reasonable uncertainties in the cost estimates.

ITEM I.A.4.2(1): RESEARCH ON TRAINING SIMULATORS

This item was evaluated in Item I.A.4.2 above and was determined to be a high priority issue. However, following the publication of NUREG-0985, Revision 1,[651] this item is now covered in Section 3.2 of the HFPP.

ITEM I.A.4.2(2): UPGRADE TRAINING SIMULATOR STANDARDS

This item was evaluated in Item I.A.4.2 above and was determined to be RESOLVED with the issuance of Regulatory Guide 1.149[439] and new requirements were established.

ITEM I.A.4.2(3): REGULATORY GUIDE ON TRAINING SIMULATORS

This item was evaluated in Item I.A.4.2 above and was determined to be RESOLVED with the issuance of Regulatory Guide 1.149[439] and new requirements were established.

ITEM I.A.4.2(4): REVIEW SIMULATORS FOR CONFORMANCE TO CRITERIA

This item was evaluated in Item I.A.4.2 above and was determined to be a high priority issue. However, following the publication of NUREG-0985, Revision 1,[651] this item is now covered in Sections 3.2 and 3.3 of the HFPP.

ITEM I.A.4.3: FEASIBILITY STUDY OF PROCUREMENT OF NRC TRAINING SIMULATOR

DESCRIPTION

The description of this safety issue in NUREG-0660[48] is as follows:

> "In addition to the increased use of industry simulators for training of NRC staff (notably, the work by OIE with the TVA training center simulators), a feasibility study of the lease or procurement of one or more simulators to be located in the NRC headquarters area will be performed. These simulators would be used in familiarizing the NRC staff with reactor operations, in assessing the effectiveness of operating and emergency procedures and in gathering data on operator performance. The study will include development of specifications, development of procurement and commissioning schedules, estimation of costs, and comparison with other methods of providing such training for NRC personnel."

Technical studies[262,263,264] that have been performed by BNL on this issue have indicated that existing simulators have significant modelling limitations. It was established that the capability of existing simulators was not acceptable at any but near-normal operating conditions, and that the lack of technical capability during two-phase conditions was significant. These results have an adverse effect on the feasibility of a training simulator for the NRC staff.

The intent of this issue is to improve the NRC staff's familiarization with reactor operations. The study is an effort to establish the feasibility of procuring an NRC training simulator. The resolution of this issue has no direct bearing on any public risk reduction and, therefore, it is concluded that this issue is a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

ITEM I.A.4.4: FEASIBILITY STUDY OF NRC ENGINEERING COMPUTER

DESCRIPTION

The description of this safety issue in NUREG-0660[48] is as follows:

> "The purpose of this study is to fully evaluate the potential value of and, if warranted, propose development of an engineering computer that realistically models PWR and BWR plant behavior for small break LOCA and other non-LOCA accidents and transients that may call for operator actions. Final development of the proposed engineering computer will depend on a number of research efforts. Risk assessment tasks (interim reliability evaluation program, or IREP, for example) to define accident sequences covering severe core damage will also provide the guidelines for the experimental and analytical research programs needed to improve the diagnostics and general knowledge of nuclear power plant systems. The programs will assist the development and testing of fast running computer codes used to predict realistic system behavior for these multiple accident studies. These codes will provide the basic models for use in the improved engineering computer as well as the capability for NRC audit of NSSS analyses."

The current status of this issue is that a report on the review of PWR simulators was completed and issued by BNL.[262] A final report on BWR simulators was also completed by BNL.[263]

Work on Plant Analyzers is continuing at BNL, INEL, and LASL. The RES staff believes that the Plant Analyzers (Engineering Computer) will be helpful in uncovering potential operational safety problems in LWRs, caused by operator errors or equipment malfunctions, which will lead to risk reductions through increased operator awareness, improved procedures, and equipment redundancy.

The Plant Analyzer is not a design tool but rather an aid to the NRC staff in performing an audit function in the licensing process. Thus, this issue will not result in a direct reduction in public risk and, therefore, is considered a licensing issue.

CONCLUSION

This item is a Licensing Issue.

REFERENCES

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

262. BNL/NUREG-28955, "PWR Training Simulator and Evaluation of the Thermal-Hydraulic Models for Its Main Steam Supply System," Brookhaven National Laboratory, 1981.

263. BNL/NUREG-29815, "BWR Training Simulator and Evaluation of the Thermal-Hydraulic Models for Its Main Steam Supply System," Brookhaven National Laboratory, 1981.

264. BNL/NUREG-30602, "A PWR Training Simulator Comparison with RETRAN for a Reactor Trip from Full Power," Brookhaven National Laboratory, 1981.

299. NUREG/CR-1482, "Nuclear Power Plant Simulators: Their Use in Operator Training and Requalification," U.S. Nuclear Regulatory Commission, August 1980.

300. NUREG/CR-2353, "Specification and Verification of Nuclear Power Plant Training Simulator Response Characteristics," U.S. Nuclear Regulatory Commission, 1982.

416. NUREG/CR-1908, "Criteria for Safety-Related Nuclear Power Plant Operator Actions: Intitial Pressurized Water Reactor (PWR) Simulator Exercises," U.S. Nuclear Regulatory Commission, September 1981.

417. NUREG/CR-2598, "Nuclear Power Plant Control Room Task Analysis: Pilot Study for Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, July 1982.

418. NUREG/CR-2534, "Criteria for Safety-Related Nuclear Power Plant Operator Actions: Initial Boiling Water Reactor (BWR) Simulated Exercises," U.S. Nuclear Regulatory Commission, November 1982.

419. NUREG/CR-3092, "Criteria for Safety-Related Nuclear Power Plant Operator Actions: Initial Simulator to Field Data Calibration," U.S. Nuclear Regulatory Commission, February 1983.

439. Regulatory Guide 1.149, "Nuclear Power Plant Simulators for Use in Operator Training," U.S. Nuclear Regulatory Commission, April 1981.

651. NUREG-0985, Revision 1, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," U.S. Nuclear Regulatory Commission, September 1984.

653. NUREG/CR-3123, "Criteria for Safety-Related Nuclear Power Plant Operator Actions: 1982 Pressurized Water Reactor (PWR) Simulator Exercises," U.S. Nuclear Regulatory Commission, June 1983.

TASK I.B: SUPPORT PERSONNEL

TASK I.B.1: MANAGEMENT FOR OPERATIONS

The objectives of this task are as follows:

(1) To improve licensee safety performance and ability to respond to accidents by upgrading the licensee groups responsible for radiation protection and plant operation in such areas as staff size; education and experience of staff members; plant operating and emergency procedures; management awareness of, and attention to, safety matters; and numbers and types of personnel available to respond to accidents.

(2) To improve licensee safety performance by establishing a full-time, dedicated, onsite safety engineering staff and providing, along with the concurrent dissemination of information to plant personnel, an integrated program for the systematic review of operating experience.

ITEM I.B.1.1: ORGANIZATION AND MANAGEMENT LONG-TERM IMPROVEMENTS

DESCRIPTION

Historical Background

This issue[48] deals with implementation of long-term organization and management improvements. The overall objective of this item is to "improve licensee safety performance and ability to respond to accidents by upgrading licensee groups responsible for radiation protection and plant operation. The areas to be upgraded include: (1) staff size; (2) education and experience of staff members; (3) plant operating and emergency procedures; (4) management awareness of, and attention to, safety matters; and (5) numbers and types of personnel available to respond to accidents." The evaluation of this issue includes the consideration of Item II.J.3.1.

To assess this safety issue, SPEB consulted with PNL as well as with NRR and RES personnel working on developing the management, organization, and staffing regulatory positions. The PNL personnel have expertise in general management, utility and nuclear plant management, reactor operations, reactor operation licensing, and general reactor safety areas. The technical analysis for this issue was provided by PNL.[64]

Safety Significance

The safety significance of this issue is the potential for accidents resulting from some measure of human error in operating a nuclear plant that may be avoidable by the resolution of this issue.

Possible Solutions

Proper management and organization will improve administration, control, prevention, coordination both within and among all key organizational components of the plant, including those located offsite. The management involved and their staff will be better qualified and trained and the staff will be increased. The management and organization will be better-prepared for both normal operations and emergency situations.

Resolution of this safety issue is assumed to involve the following:

(1) Each utility (licensee/applicant) will be required to submit a new proposed organization and management plan which will be reviewed by the NRC, including a site review. No additional management staff will be required, but the qualifications and training of the management staff and the organization effectiveness will be improved substantially at most plants.

(2) Up to 14 additional people will be required to be added to the staff depending on the plant. These people will be maintenance (~9), health physics and chemistry (~3) and training (~2) personnel. Not included are staff to man a plant-specific simulator, if required by the NRC (this was considered under Item I.A.4.1).

It is anticipated that 25% of the plants will require no staff additions, 50% will require only 8 people, and 25% will require all 14 people. Thus, on the average, a plant would require 7 additional staff members.

(3) The OIE staff at NRC will perform annual assessments to assure each utility is satisfactorily meeting NRC management and organization requirements as identified in the initial plant review.

(4) Regulatory Guides 1.33[225] and 1.8[226] will be revised and issued, along with other appropriate regulatory guidance, to define requirements in this area.

(5) Implementation of this safety issue at all operating plants and for plants applying for an operating license is assumed to begin in FY 1984 with all plants covered by mid-FY 1985. This includes annual followup assessments under way in FY 1985.

PRIORITY DETERMINATION

Assumptions

The major benefit from resolution of this safety issue will be reduction in human errors (operators and maintenance personnel) resulting in lower public risk. This applies to the remaining operating life of all nuclear power plants (142), currently operating and under construction, subsequent to implementation of the solution in 1985, which is approximately 26 years.

The PNL staff estimated that the proper actions could potentially result in a 20% reduction in human errors at a nuclear plant. However, many of the plants (assumed to be 25%) are already well-managed and organized. These would see no further improvement. Another 50% would obtain only half the benefit and the remaining 25% would obtain the full benefit. An average value of 10% for reduction of human errors is anticipated for the nuclear industry at large.

Frequency Estimate

All accident sequences, except an interfacing system LOCA, would be affected. Reducing the human error rate by 10% is calculated to decrease the frequency of core-melt in Oconee by 5×10^{-6}/plant-yr. The frequency of core-melt in Grand Gulf was assumed to be reduced by the same ratio, or 2×10^{-6}/plant-yr.

Consequence Estimate

All release categories are affected and the reduction in public risk is estimated to be 13 man-rem/plant-yr for PWRs and 15 man-rem/plant-yr for BWRs, based on the WASH-1600[16] estimate of release and assuming a typical midwest-type meteorology and an average population density of U.S. reactor sites of 340 people per square mile. Assuming 94 PWRs and 48 BWRs with an average remaining life of 26 years after this issue is implemented in 1985, the total public risk reduction is 50,400 man-rem.

Cost Estimate

Industry Cost: The major cost of resolving this safety issue is that associated with possible additional staffing required at a nuclear plant. Both BWRs and PWRs would be affected equally. Specifically, industry costs associated with this issue are expected to be as follows:

(1) An average of 7 people per plant is used in the calculation of industry labor for operation and maintenance.

(2) Approximately 2 man-years of effort for "intermediate case" plants would be required for preparing the initial management plan and reviewing it with the NRC. (Triple that for "worst case" plants and half that for "best case" plants). An average of 2.75 man-years per plant is used in the calculation of industry labor for implementation.

(3) Approximately 1 man-month of utility effort would be required at each plant in supporting the annual NRC management assessment of the solution.

The total industry costs calculated by PNL[64] were \$33M for implementation and \$2.27M for operation and maintenance.

NRC Cost: NRC costs associated with resolving this safety issue are expected to be as follows:

(1) Approximately 22 man-years of effort by NRR and RES to develop the long-term regulatory position on management and organization after FY-1982.

(2) Approximately 2 man-years to write, obtain, and issue comments on revised and new regulatory guides. The major development effort behind these guides is included in (1) above.

(3) Approximately 5 man-months to review the initial management and organization plan proposed for each plant. This includes time for the site visit and assessment report.

(4) Approximately 0.5 man-months to perform an annual assessment of the solution at each plant.

The total NRC cost calculated[64] by PNL was approximately \$30.8M.

Value/Impact Assessment

Based on the total public risk reduction of 50,400 man-rem, the value/impact score is given by:

$$S = \frac{50,400 \text{ man-rem}}{\$(33 + 2.27 + 30.8)\text{M}}$$

$$= 763 \text{ man-rem/\$M}$$

Other Considerations

There would be some reduction in occupational risk primarily from lowering occupational exposure due to fewer unplanned outages caused by human error. Maintenance staffs are primarily impacted; however, both operating and maintenance staffs will benefit from avoidance of major accidents.

The potential for exposure reduction is expected to be about 10% for those 25% of "worst case" plants, half that for the 50% of "intermediate case" plants, and none for the 25% of "best case" plants. An average value of 5% is used in the calculations which follow. It is estimated that 300 to 500 man-rem of occupational exposure occur annually at a typical facility. If we assume 400 man-rem as a best estimate, the 5% reduction results in an occupational dose reduction of 20 man-rem/plant-yr. For 142 plants with an average remaining lifetime of approximately 26 years, the total occupational risk reduction from this source is approximately 75,000 man-rem.

The industry accident avoidance cost was estimated by PNL[64] to be \$26.2M.

CONCLUSION

The potential public risk reduction is relatively large (50,400 man-rem) and the potential for occupational risk reduction is also large (75,000 man-rem), if the estimate of the reduction in human error is correct. Since most of the costs are due to additional utility staff, this value/impact could be higher if a resolution were found that did not require added staff. Therefore, based on the large potential risk reduction, this issue was given a MEDIUM priority ranking.

ITEM I.B.1.1(1): PREPARE DRAFT CRITERIA

This item was evaluated in Item I.B.1.1 above and was determined to be a medium priority issue. However, following the publication of NUREG-0985 Revision 1,[651] this item is now covered in Sections 6.1 and 6.2 of the HFPP.

ITEM I.B.1.1(2): PREPARE COMMISSION PAPER

This item was evaluated in Item I.B.1.1 above and was determined to be a medium priority issue. However, following the publication of NUREG-0985, Revision 1,[651] this item is now covered in Sections 6.1 and 6.2 of the HFPP.

ITEM I.B.1.1(3): ISSUE REQUIREMENTS FOR THE UPGRADING OF MANAGEMENT AND TECHNICAL RESOURCES

This item was evaluated in Item I.B.1.1 above and was determined to be a medium priority issue. However, following the publication of NUREG-0985, Revision 1,[651] this item is now covered in Sections 6.1 and 6.2 of the HFPP.

ITEM I.B.1.1(4): REVIEW RESPONSES TO DETERMINE ACCEPTABILITY

This item was evaluated in Item I.B.1.1 above and was determined to be a medium priority issue. However, following the publication of NUREG-0985, Revision 1,[651] this item is now covered in Sections 6.1 and 6.2 of the HFPP.

ITEM I.B.1.1(5): REVIEW IMPLEMENTATION OF THE UPGRADING ACTIVITIES

OIE routinely develops and issues inspection procedures which address new or revised regulations and requirements.[441] Thus, this item has been RESOLVED and no new requirements were established.

ITEM I.B.1.1(6): PREPARE REVISIONS TO REGULATORY GUIDES 1.33 AND 1.8

This item was evaluated in Item I.B.1.1 above and was determined to be a medium priority issue. However, following the publication of NUREG-0985, Revision 1,[651] the revision to Regulatory Guide 1.8[226] is now covered in Sections 1.2 and 2.1 of the HFPP. The revision to Regulatory Guide 1.33[225] is now covered[652] in Subtask 1 of the work plan for Issue 75.

ITEM I.B.1.1(7): ISSUE REGULATORY GUIDES 1.33 AND 1.8

This item was evaluated in Item I.B.1.1 above and was determined to be a medium priority issue. However, following the publication of NUREG-0985, Revision 1,[651] the revision to Regulatory Guide 1.8[226] is now covered in Sections 1.2 and 2.1 of the HFPP. The revision to Regulatory Guide 1.33[225] is now covered[652] in Subtask 1 of the work plan for Issue 75.

ITEM I.B.1.3: LOSS OF SAFETY FUNCTION

DESCRIPTION

Historical Background

This TMI Action Plan[48] item concerns regulatory action at an operating nuclear power plant in the event of human error leading to complete loss of a safety function required by the plant's Technical Specifications. The following three options specified in the TMI Action Plan[48] were considered:

1. Require licensees to immediately place the plant in the safest shutdown cooling condition following a total loss of a safety function due to personel error if a total loss of a safety function had occurred within the previous year or two. Resumption of operation would require NRC approval based on a review of the licensee's program for corrective action.
2. Use existing enforcement options (citations, fines, shutdowns).
3. Use approaches such as a point system, licensee probations, and (in the extreme) license revocations.

Safety Significance

Loss of a required safety function can lead to an increase in the probability that an event with an accident-initiating potential, should it occur, would lead to an actual major accident. This probability increase could be more or less substantial, depending on the specific function lost. The safety concern is heightened when the loss of safety function is caused by human error and this occurs more than once in a year or two. Such repeated personnel failures can bring into question whether the reliability of safety-related personnel actions at the plant involved are generally up to the standards expected and assumed in safety evaluations.

Solution

Option 2 was selected as the best option that provides the latitude needed by NRC for determination whether a particular event falls under the definition of a "loss of safety function," the role of human error in causing the event, the acuteness of the risk, the urgency and nature of appropriate remedial action, conditions for resumption of operation, and such considerations as the public health-and-safety need for power at the time.[265,266,267,287,288]

With the selection of Option 2, Item I.B.1.3 of the TMI Action Plan was terminated as such, having become part of the Enforcement Policy issue (Item IV.A.2) which has been completed.[288] This item is related to improving the NRC capability to make independent assessments of safety and, therefore, is considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

ITEM I.B.1.3(1): REQUIRE LICENSEES TO PLACE PLANT IN SAFEST SHUTDOWN COOLING FOLLOWING A LOSS OF SAFETY FUNCTION DUE TO PERSONNEL ERROR

This Licensing Issue was evaluated in Item I.B.1.3 above and was determined to be resolved.

ITEM I.B.1.3(2): USE EXISTING ENFORCEMENT OPTIONS TO ACCOMPLISH SAFEST SHUTDOWN COOLING

This Licensing Issue evaluated in Item I.B.1.3 above and was determined to be resolved.

ITEM I.B.1.3(3): USE NON-FISCAL APPROACHES TO ACCOMPLISH SAFEST SHUTDOWN COOLING

This Licensing Issue was evaluated in Item I.B.1.3 above and was determined to be resolved.

REFERENCES

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

225. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," U.S. Nuclear Regulatory Commission, February 1978.

226. Regulatory Guide 1.8, "Personnel Selection and Training," U.S. Nuclear Regulatory Commission, May 1977.

265. Memorandum for the Commissioners from W. Dircks, "Enforcement Policy," March 18, 1980.

266. SECY-80-139A, "NRC Enforcement Program," August 27, 1980.

267. Memorandum for R. Purple from R. Minogue, "TMI Action Plan," October 24, 1980.

287. SECY-81-600A, "Revised General Statement of Policy and Procedure for Enforcement Actions," December 14, 1981.

288. Federal Register, Vol. 47, pp. 9987-9995 "General Statement of Policy and Procedure for Enforcement Actions," March 9, 1982.

441. Memorandum for H. Denton from R. DeYoung, "Commission Paper on the Prioritization of Generic Safety Issues," April 20, 1983.

651. NUREG-0985, Revision 1, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," U.S. Nuclear Regulatory Commission, September 1984.

652. Memorandum for W. Dircks from R. DeYoung, "Elimination of Duplicative Tracking Requirements for Revision of Regulatory Guide 1.33," July 26, 1984.

TASK I.C: OPERATING PROCEDURES

The objective of this task is to improve the quality of procedures to provide greater assurance that operator and staff actions are technically correct, explicit, and easily understood for normal, transient, and accident conditions. The overall content, wording, and format of procedures that affect plant operation, administration, maintenance, testing, and surveillance will be included.

ITEM I.C.1: SHORT-TERM ACCIDENT ANALYSIS AND PROCEDURES REVISION

ITEM I.C.1(4): CONFIRMATORY ANALYSES OF SELECTED TRANSIENTS

DESCRIPTION

Historical Background

This TMI Action Plan item[48] requires confirmatory analyses of selected transients by NRR to provide the basis for comparisons with analytical methods being used by the reactor vendors. These comparisons will assure the adequacy of the analytical methods being used to generate emergency procedures. NRC has performed a limited number of confirmatory transient analyses. The rest are currently being defined.

Safety Significance

The safety significance is the reduction in operator errors and upgrading of operating systems through confirmatory analyses of selected transients by NRC/NRR. These confirmatory analyses should provide greater assurance that operator and staff actions are technically correct.

Possible Solution

Confirmatory analyses, using the best available computer codes, will provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons, together with comparisons to other data, will constitute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures.

PRIORITY DETERMINATION

Frequency Estimate

To evaluate this issue, PNL assumed[64] improvements in two areas. The reduction in human error rate for operators was estimated to be 7%. Other operation improvements (set points for control systems, maintenance, hardware upgrade, etc.) were estimated at 4.5%. The total improvement percentages were applied to the base-case frequencies and affected release categories for both PWR and BWR type plants. The dominant accident sequences and base-case frequencies for the Oconee

(B&W) plant were used for the PWR plants. For BWR plants, Grand Gulf 1 was used as the model.

For PWR plants the base-case core-melt frequency was determined to be 8.2×10^{-5}/RY. The adjusted core-melt frequency, considering the above improvements, was determined to be 7.3×10^{-5}/RY. The result was a reduction in core-melt frequency of 9×10^{-6}/RY. For BWR plants the base-case and adjusted core-melt frequencies were determined to be 3.7×10^{-5}/RY and 3.3×10^{-5}/RY, respectively. The reduction in core-melt frequency for BWR plants was 4×10^{-6}/RY.

Consequence Estimate

Because of the multifactor influence of the estimated improvements, all seven of the PWR release categories and all four of the BWR release categories were assumed to be affected. The potential public risk reduction for PWR plants was calculated to be 6.5×10^4 man-rem, assuming 95 plants with an average remaining life of 28.5 years. The potential public risk reduction for BWR plants was calculated to be 4×10^4 man-rem, assuming 49 plants with an average remaining life of 27 years. In all cases a population density of 340 persons per square mile and typical meteorology were assumed. The total reduction in risk to the public, based on the above results, was about 1.05×10^5 man-rem.

Cost Estimate

The industry cost was estimated at $61M. This estimate included: (1) an industry rate of $1900/man-week, (2) 30 man-weeks to implement the resolution, (3) 7 man-weeks per reactor year for operation and maintenance, (4) 144 plants, and (5) an average remaining life of 28 years. The NRC cost including implementation and reviews was estimated at $2.8M. The total industry and NRC cost was therefore estimated at approximately $64M.

Value/Impact Assessment

Based on a total public risk reduction of 1.05×10^5 man-rem and a total cost of $64M, the value/impact score is given by:

$$S = \frac{1.05 \times 10^5 \text{ man-rem}}{\$64M}$$

$$= 1{,}650 \text{ man-rem/\$M}$$

Other Considerations

Other factors which have been considered are the accident avoidance costs and the potential occupational risk reductions. The accident avoidance cost is the product of the reduction in the probability of core-melt and industry cost factors, assuming cleanup, repair, refurbishment, and replacement power cost over a 10-year period.

The total accident avoidance costs for all PWRs (95) and all BWRs (49), which includes current operating plants and those plants expected to commence operation, are estimated to be approximately $49M. Therefore, the net industry cost for this issue, when reduced by the accident avoidance costs, would be approximately $12M.

The occupational dose incurred from accident recovery is estimated at 20,000 man-rem[64]. The total occupational dose reduction due to accident avoidance, considering all PWRs and BWRs, is 600 man-rem. If we assume a 5% reduction in annual operational doses due to imposed operating guidelines and upgraded control systems, the best estimate annual operational dose reduction is 20 man-rem/RY. For all plants and all remaining plant life, the potential occupational dose reduction is 81,000 man-rem. The above estimates indicate that the potential reduction in occupation doses during normal operation is significant and further supports a high priority ranking for this issue.

CONCLUSION

Based on the value/impact score and the potential reduction in core-melt frequency, the issue would be classified as medium priority, but, because of the total public risk reduction of 105,000 man-rem, the issue was given a high priority ranking. However, since the prioritization of this issue, all required work was completed,[382,383] the issue was RESOLVED, and no new requirements were established.

ITEM I.C.9: LONG-TERM PROGRAM PLAN FOR UPGRADING OF PROCEDURES

DESCRIPTION

Historical Background

The NRC effort for this TMI Action item[48] (to be led by NRR but to involve IE, SD, and RES) was to develop a long-term program plan for the upgrading of plant procedures. This plan would incorporate and expand on current efforts associated with the development, review, and monitoring of procedures. Consideration of studies to ensure clear procedures with particular emphasis on diagnostic aids for off-normal conditions were called for. The interrelationships of administrative, operating, maintenance, test, and surveillance procedures were to be considered. The topics of emergency procedures, reliability analysis, human factors engineering, crisis management, and operator training were also to be addressed.

That part of Item I.C.9 related to emergency operating procedures (EOP), has been implemented in accordance with Item I.C.1 of NUREG-0737.[98] In regard to the EOPs, SECY-82-111[151] requested Commission approval of a set of basic requirements for emergency response capability and approval for the staff to work with licensees to develop plant specific implementation schedules. A significnt amount of work on emergency operating procedures has been completed. All four NSSS vendors have submitted technical guidelines based on re-analysis of accidents and transients. These are in the final stages of review. In the area of human factors, a survey of current practices, research on EOPs, and pilot monitoring of some Near-Term Operating License (NTOL) plants have been completed and criteria for development of emergency operating procedures were published for public comment in NUREG-0799.[191] NUREG-0899[192] was published in final form in September 1982 and incorporated resolution of comments received on NUREG-0799.[191]

The recommended requirements for EOPs,[151] which include some of these completed or nearly completed tasks, have been conditionally approved.[190]

That part of Item 1.C.9 [48] pertaining to other long-term procedures (which were not addressed in NUREG 0737[98]) require further staff effort. The priority ranking for this remaining staff effort is discussed herein.

Safety Significance

Resolution of this issue is expected to have a significant impact on plant procedures. The changes in procedures are in turn expected to improve the safety-related performance of all plant operations staff. This would apply to both routine and abnormal operating conditions.

Possible Solution

Staff actions under Item I.C.9[48] which pertain to normal and abnormal operating procedures, maintenance, test, surveillance, and other safety-related procedures are continuing. The staff effort related to the above is scheduled in three phases:

- (a) Survey ongoing studies, existing procedures, and practices of related industries; assess problems; and prioritize solutions (FY 1982-1983).
- (b) Prepare guidance (NUREGs, Regulatory Guides) for industry use (FY 1983-1984).
- (c) Issue requirements, prepare inspection guidance, review or audit as necessary (FY 1985-1986).

PRIORITY DETERMINATION

Frequency Estimate

To estimate the change in core-melt frequency for this issue, PNL[64] assumed a human error rate reduction of 30% for operations staff. PNL also assumed that the dominant accident sequences for the Oconee (B&W) plant were representative of all PWRs, and that the fractional risk and core-melt frequency reductions were applicable to the representative BWR (Grand Gulf).

For PWRs, the base-case core-melt frequency was determined to be 7.8×10^{-5}/RY. The adjusted core-melt frequency, considering the above improvement, was determined to be 5.6×10^{-5}/RY. The result was a reduction in core-melt frequency of 2.2×10^{-5}/RY for PWRs. In the case of the BWRs, the base-case core-melt frequency was determined to be 3.5×10^{-5}/RY. The reduction in core-melt frequency for BWRs was 9.9×10^{-6}/RY.

Consequence Estimate

All seven of the PWR release categories and all four of the BWR release categories were affected by this improvement. The potential public risk reduction for PWRs was calculated to be 53 man-rem/RY, assuming WASH-1400[16] release categories, a population density of 340 persons per square mile, and typical meteorology. The reduction in public risk for the BWRs was calculated to be 64 man-rem/RY.

The total public risk reduction for all plants (90 PWRs and 44 PWRs) was 2.1×10^5 man-rem, assuming an average remaining life of 28 years.

Cost Estimate

The industry costs were estimated at \$447M. This included \$67M to implement and upgrade, and \$380M due to operation and maintenance. The NRC cost including implementation and reviews was estimated at \$9M. The total industry and NRC cost was therefore estimated at approximately \$456M.

Value/Impact Assessment

Based on a total public risk reduction of 2.1×10^5 man-rem the value/impact score is given by

$$S = \frac{2.1 \times 10^5 \text{ man-rem}}{\$456M}$$

$$= 461 \text{ man-rem/\$M}$$

Other Considerations

In the analysis of this issue, PNL[64] assumed a uniform 30% improvement in human error, including maintenance, through the dominant accident sequences. The 30% improvement is expected to over-estimate reductions in maintenance outages. It is assumed that no significant reductions in maintenance outages would reduce the potential risk reduction calculated by PNL approximately 10%. These improvements transcended normal, abnormal, and emergency procedures during the event sequences as described in NUREG-0660,[48] Item I.C.9. However, the EOP concerns originally included in Item I.C.9 were separated out of Item I.C.9, and addressed in NUREG-0737.[98] Based on subsequent discussions between the NRC staff and the PNL analyst, it was agreed that the results of the dominant accident sequences would be strongly influenced by the EOPs. This situation is expected to result in little or no change to the above calculated value/impact assessment score of 461 man-rem/\$M. The reason being that the smaller risk reduction that can be attributed to this issue, after the EOP effect is removed, is balanced by lower implementation cost to complete the remaining part of this issue. The beneficial reduction in core-melt frequency and public risk calculated in the PNL analysis is however significantly less when dominant effects of the improvements in the EOPs are removed from this issue. If we assume that improved EOPs will contribute approximately 75% toward reducing the core-melt frequency, and public risk, the benefit (risk reduction) attributed to improvements and upgrading of the other procedures is 25% of the total benefits previously calculated. This results in a total reduction in public risk of $(0.9)(0.25)(2.1 \times 10^5)$ man-rem or 47,000 man-rem. These reductions are attributable to that part of Item I C.9 not addressed in Item I.C.1 of NUREG-0737.[98]

CONCLUSION

With the exclusion of the EOPs (which are implemented in NUREG-0737[98]), this issue was given a medium priority ranking. However, with the publication of NUREG-0985, Revision 1,[651] the item is now covered in Section 4.1 of the HFPP.

REFERENCES

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

98. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980.

151. SECY-82-111, "Requirements for Emergency Response Capability," March 11, 1982.

190. Memorandum for W. Dircks from S. Chilk, "Staff Requirements-Affirmative Session, 11:50 a.m., Friday July 16, 1982," July 20, 1982.

191. NUREG-0799, "Draft Criteria for Preparation of Emergency Operating Procedures," U.S. Nuclear Regulatory Commission, July 1, 1982.

192. NUREG-0899, "Guidelines for Preparation of Emergency Operating Procedures - Resolution of Comments on NUREG-0799," U.S. Nuclear Regulatory Commission, September 3, 1982.

382. Memorandum for W. Minners from R. Mattson, "Schedules for Resolving and Completing Generic Issues," January 21, 1983.

383. Memorandum for W. Dircks from R. Mattson, "Closeout of TMI Action Plan I.C.1(4), Confirmatory Analyses of Selected Transients," November 12, 1982.

651. NUREG-0985, Revision 1, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," U.S. Nuclear Regulatory Commission, September 1984.

TASK I.D: CONTROL ROOM DESIGN

The objective of this task is to improve the ability of nuclear power plant control room operators to prevent accidents or cope with accidents if they occur by improving the information provided to them.

ITEM I.D.3: SAFETY SYSTEM STATUS MONITORING

DESCRIPTION

Historical Background

This TMI Action Plan item[48] recommends that a study be undertaken to determine the need for all licensees and applicants not committed to Regulatory Guide 1.47[150] to install a bypass and inoperable status indication system or similar system.

Safety Significance

Implementation of a well-engineered bypass and inoperable status indication system could provide the operator with timely information on the status of the plant safety systems. This operator aid could help eliminate operator errors such as those resulting from valve misalignment due to maintenance or testing errors.

Possible Solutions

A study of current industry (nuclear and others) practices could be undertaken to evaluate possible methods/systems for verifying correct system alignment. In conjunction with this, a study of failures of systems due to pump or valve unavailability could be undertaken. Based on the results, a requirement to backfit or not backfit Regulatory Guide 1.47[150] (or a revision thereof) would be set forth.

PRIORITY DETERMINATION

Assumptions

If the system is integrated with the overall control room, then it could be expected that it would reduce operator error, which in turn will lower the risk associated with operation of the monitored safety systems.

For some utilities this "new" system may result in a modest but significant reduction in operator error during an emergency whereas in others the system may have no discernible effect. An average of about 2% was applied to all presently operating plants. Plants not yet licensed or undergoing licensing are committed to Regulatory Guide 1.47.[150]

In an analysis of this issue performed by PNL,[64] Oconee 3 was selected as the representative PWR. It was assumed that the fractional risk and core-melt frequency reductions for a representative BWR (Grand Gulf 1) will be equivalent to those calculated for the representative PWR.

Frequency/Consequence Estimate

The reduction in core-melt frequency ($\bar{\Delta}F$) for Oconee was calculated to be 8.7×10^{-7}/plant-yr, based on adjustment to the risk equation parameters affected by issue resolution and then a calculation of a core-melt frequency and comparison to the base core-melt frequency.

Based on a scaling calculation (see NUREG/CR-2800[64]), the frequency reduction ($\bar{\Delta}F$) for Grand Gulf was 3.9×10^{-7}/plant-yr. The reduction in public risk was calculated (assuming WASH-1400[16] release categories, typical midwest-site meteorology, and a uniform population density of 340 people per square-mile) to be 5.9 man-rem/plant-yr for Oconee and 7.1 man-rem/plant-yr for Grand Gulf.

The total risk reduction for this issue was calculated to be 1.2×10^4 man-rem, based on 5.9 man-rem/plant-yr for 47 PWRs, 7.1 man-rem/plant-yr for 24 BWRs, and average remaining lives of 28 years and 25 years for PWRs and BWRs, respectively.

Cost Estimate

Industry Cost: Installation costs (including labor and equipment) were estimated as follows:

	Equipment	Cost
(a)	Cable 30 miles @ $6.00/100 Lft	$ 9,500
(b)	Elec. Penetration Limitations	300,000
(c)	Cable tray and additional termination	10,000
(d)	Intermediate Logic Panel	100,000
(e)	Control Room Alarms/Indications	10,000
	Total:	$429,500

	Other	Cost
(a)	Design labor @ 12 man-months	$ 75,000
(b)	Installation Labor = 17 man-months	100,000
(c)	QA	40,000
	Total:	$215,000

Therefore, the total implementation cost to industry is $644,500/plant.

Maintenance of the solution by industry is estimated to require 1 man-week/plant. At a cost of $1,000/plant-yr, this amounts to a total industry cost of $1.9M. Therefore, the total industry cost is $48M.

NRC Cost: NRC labor for development of the resolution is estimated to be 0.5 man-year. Review and implementation of the solution is estimated to take 4 man-weeks/plant. Therefore, the total NRC cost is $0.6M.

Value/Impact Assessment

Based on a public risk reduction of 1.2×10^4 man-rem, the value/impact score is given by:

$$S = \frac{1.2 \times 10^4 \text{ man-rem}}{\$(48 + 0.6)M}$$

$$= 240 \text{ man-rem/\$M}$$

Uncertainty

Because the estimate of the value/impact score relies heavily on the estimated value of the possible reduction in human error, there may be wide variance in the effective improvement.

Additional Considerations

(1) To resolve this issue effectively, it should be done in conjunction with Item I.D.1 which addresses control room design review. This issue was not explicitly included in the present Commission requirement for Control Room Design (Item I.D.1) which is to be implemented in accordance with SECY-82-111[151] and a letter[376] issued to licensees of all operating plants.

(2) As another potentially significant consideration, resolution of this issue may provide a reduction in safety system unavailability due to the contribution of maintenance and testing.

(3) DHFS is presently contracting with various groups to study this issue.[152,153] These studies could better define the assumptions (for risk reduction) used in the calculation. This would then provide better data for a benefit/cost study to determine implementation.

CONCLUSION

Based on the estimated public risk reduction and the value/impact score, this issue was given a medium priority ranking. However, with the publication of NUREG-0985, Revision 1,[651] this item is now covered in Section 5.2 of the HFPP.

ITEM I.D.4: CONTROL ROOM DESIGN STANDARD

DESCRIPTION

Historical Background

This issue was documented in NUREG-0660[48] and emphasized a need for guidance on the design of control rooms to incorporate human factor considerations.

Safety Significance

Control rooms and control panels which incorporate human factor considerations can greatly enhance operator performance. This could contribute to a reduction in operator error and, therefore, a potential reduction in the frequency of core-melt accidents.

Possible Solution

An NRC Regulatory Guide endorsing industry standard(s) could be developed with the intention of providing: (1) guidance for the design of control rooms and, (2) the evaluation criteria for use in the licensing process.

PRIORITY DETERMINATION

Assumptions

PNL did an assessment of this issue.[64] From the representative PWR (Oconee) and BWR (Grand Gulf), those parameters in the risk equations requiring direct operator actions were considered affected. That is, it was assumed that the probability of operator error for these parameters were decreased by 3% based on resolution of this safety issue. It was assumed that only plants to be licensed beyond 1986 would be affected.

Frequency/Consequence Estimate

The affected accident sequences and associated base-case frequencies were determined. From these frequencies, the (Affected Release Categories) base case frequencies were determined and a new base case core-melt frequency was calculated. This was 3.1×10^{-5}/plant-year for the PWRs and 6.1×10^{-6}/plant-year for the BWRs. In addition, a new base case public risk was calculated for the affected parameters. This was 79.1 man-rem/plant-yr for PWRs and 40.4 man-rem/plant-yr for BWRs. To determine a change in public risk due to issue resolution, the affected parameters were adjusted by 3% and the frequencies of the associated sequences and release categories were determined. A new overall core-melt frequency was then determined. The new core-melt frequency was 3.01×10^{-5}/plant-year for PWRs and 5.95×10^{-6}/plant-yr for BWRs. Also a new public risk was then calculated: 76.9 man-rem per plant-yr for PWRs and 39.2 man-rem per plant-yr for BWRs.

From the above numbers, the reduction in core-melt frequency (due to issue resolution) was calculated to be 9×10^{-7}/plant-yr for PWRs and 1.8×10^{-7}/plant-yr for BWRs. The public risk reduction was calculated to be 2.2 man-rem/plant-yr for PWRs and 1.2 man-rem/plant-yr for BWRs. Therefore, the total public risk reduction, based on 10 PWRs and 5 BWRs and an average remaining life of 30 years, was calculated to be 840 man-rem.

Cost Estimate

Industry Cost: It was assumed that for those plants expected to be completed after 1990, the cost to implement the standard will be part of the basic cost. For those plants expected to be completed between 1987 and 1990, the cost to redesign the control room was estimated to be $100,000 per plant. This is based on the assumption that, in all likelihood, draft standards will be available and will be used and then only minor changes will be needed. Also, it is assumed that the standards will not require significant equipment additions, but only reworking of preliminary designs. Since there are about 10 plants to be completed between 1987 and 1990, total industry cost for implementation is $1M. No additional cost for yearly industry operation and maintenance was assumed.

NRC Cost: The NRC cost estimate was based on an assumed $300,000 expenditure for regulatory guide development. It was assumed that additional NRC labor of about 4 man-weeks per plant would be necessary to review the modifications that would be required for the 10 plants completed between 1987 and 1990. This equals a cost of about $9,000/plant or $90,000 total. The total NRC cost is then $390,000.

Value/Impact Assessment

Based on a total public risk reduction of 840 man-rem, the value/impact score is given by:

$$S = \frac{840 \text{ man-rem}}{\$(1 + 0.39)M}$$

$$= 600 \text{ man-rem}/\$M$$

Uncertainty

The human error reduction is not easily quantifiable. Three percent was used here but it is subject to large uncertainty.

Other Considerations

(1) The issue was assumed to affect only future plants. Present NRC guidelines in NUREG-0700[474] are to be applied to all existing plants and NTOLs.

(2) IEEE Standards are under development.

CONCLUSION

Based on the above value/impact score, this issue was given a medium priority ranking. However, with the publication of NUREG-0985, Revision 1,[651] this item is now covered in Sections 5.1 and 5.2 of the HFPP.

ITEM I.D.5: IMPROVED CONTROL ROOM INSTRUMENTATION RESEARCH

ITEM I.D.5(1): OPERATOR-PROCESS COMMUNICATION

DESCRIPTION

Historical Background

This issue was documented in the TMI Action Plan[48] and focused on the need to evaluate the operator machine interface in reactor control rooms. The emphasis of this portion of the overall issue was the use of lights, alarms, and annunciators.

Safety Significance

The method of presentation of information can significantly enhance the performance of the control room operators and thereby potentially affect operator error.

Possible Solution

It was proposed that current practice and use of lights, alarms, and annunciators be reviewed to assess how well they facilitate operator-machine interaction and minimize errors. RES has studied the area of control room alarms and annunciators (through a contractor) and the results were reported in NUREG/CR-2147.[244] Based on this report, RES issued a Research Information Letter[245] (RIL-124) which provided a recommendation for further action.

CONCLUSION

This item was RESOLVED and no new requirements were established.

ITEM I.D.5(2): PLANT STATUS AND POSTACCIDENT MONITORING

DESCRIPTION

Historical Background

This issue was documented in the TMI Action Plan[48] and focused on the need to improve the ability of reactor operators to prevent, diagnose, and properly respond to accidents. The emphasis was on the information needs (i.e., indication of plant status) of the operator.

Safety Significance

In order for the operators to perform their functions it is necessary that they receive all the necessary information on the plant status. This can enhance operator performance (and therefore reduce operator error).

Possible Solution

Accident sequences should be analyzed to determine the information required to provide unambiguous indication of plant status. Specific instrumentation and ESF status monitoring needs would then be determined. PWR instrumentation requirements were analyzed in NUREG/CR-1440[241] and BWR instrumentation requirements were analyzed in NUREG/CR-2100.[242] ESF Status Monitoring requirements were also studied in NUREG/CR-2278.[243] Research Information Letter (RIL) No. 98[246] was issued in August 1980. This RIL transmitted "the results of completed research describing an improved method for analyzing accident sequences." Revision 2 to Regulatory Guide 1.97[55] was issued in December 1980. (See also Item II.F.3, "Instrumentation for Monitoring Accident Conditions.") Present plans include implementation of this guide at all plants.[151,376]

CONCLUSION

This item was RESOLVED and new requirements were established.

ITEM I.D.5(3): ON-LINE REACTOR SURVEILLANCE SYSTEM

DESCRIPTION

This item was documented in the TMI Action Plan[48] based on the work being performed by ORNL. A continuous on-line automated surveillance system was installed at Sequoyah-1 (PWR) and information has been obtained throughout the first fuel cycle.

The demonstration at Sequoyah is planned to continue through the second fuel cycle (mid-1984). A similar demonstration at an operating BWR is planned for initiation in 1984. The system has the potential to provide diagnostic information to predict anomalous behavior of operating reactors which could be used to maintain safe conditions.

Noise surveillance and diagnostic techniques associated with the on-line reactor surveillance system have shown their safety significance and the results of the research have been and are being used by NRC in regulatory activities as discussed below. Monitoring of neutron noise in BWRs was used to detect and monitor the impacting of instrument tubes against fuel boxes. The technique was used by NRC and its consultants to verify that partial power operation was safe until the next scheduled fuel outages for some 10 BWRs. Pressure noise surveillance was used at TMI-2 to monitor and guide degassification of the primary loop. Recently, the data obtained from the on-line surveillance demonstrated at Sequoyah-1 was used by NRC and its consultants in the assessment of loose thermal shields in Oconee Units 1, 2, and 3. In yet another example, NRR is currently using results of this research in BWR stability determinations associated with regulatory actions pertaining to Dresden.

CONCLUSION

Based on the ongoing programs, we conclude that the technical resolution of this issue has been identified.

ITEM I.D.5(4): PROCESS MONITORING INSTRUMENTATION

DESCRIPTION

This item was documented in the TMI Action Plan[48] and was to explore the feasibility of using new concepts for measuring certain reactor parameters. A directly related issue, Item II.F.2 in NUREG-0737,[98] mandated that industry develop and implement PWR liquid level detection systems. NRC evaluated a number of systems at the LOCA experiment facilities at ORNL and INEL.

CONCLUSION

This item has been RESOLVED and no new requirements were established.

ITEM I.D.5(5): DISTURBANCE ANALYSIS SYSTEMS

DESCRIPTION

Historical Background

This issue was documented in the TMI Action Plan[48] and its objective was to explore advanced disturbance analysis systems for possible application to nuclear power plants.

Safety Significance

If potential transient events could be anticipated and terminated earlier and if operator response could be enhanced, then the core-melt frequency may be reduced. Advanced disturbance analysis systems could possibly provide the capabilities to achieve this.

Possible Solution

The purpose of this item was to assess the need, feasibility, and adequacy of advanced disturbance analysis systems. EPRI is presently doing research in this area.

PRIORITY DETERMINATION

Assumptions

To evaluate this item, we assumed that the advanced disturbance analysis system would include the implementation of a continuous on-line surveillance system, as discussed in Item I.D.5(3). [A liquid level detection system was assumed available because it is already required - Items I.D.5(4) and II.F.2.]

In a PNL assessment of this issue,[64] it was decided that a risk reduction could be estimated by assuming a reduction in operator errors. Operator error was assumed to be reduced by 2% due to the implementation of this additional operator aid. Also, a reduction in the number of transients requiring shutdown was assumed based on the potential that the operators will be able to terminate some transients before the need for shutdown. Reduced transient frequencies were calculated based on a recent EPRI analysis.[307] The basis for choosing the transients was that either the detection time leading up to the transient or the time from the transient occurrence to shutdown was perceived to be longer than 30 minutes, enabling the advanced diagnostic system to diagnose the problem and provide possible solutions for the operator.

Furthermore, for purposes of this study, it was assumed that the operator could only respond with actions to 80% of the transients listed that would occur during the remaining lifetimes of the subject plants. Of the 80%, only 25% of the operator's actions was assumed to prevent the need for shutdown. The average plant shutdown was assumed to last 0.75 day. Therefore, reduction in unscheduled outages is calculated as follows:

$$\text{PWR:} \quad \left(\frac{4.63 \text{ transients}}{\text{plant-yr}}\right)(0.80)(0.25)\left(\frac{0.75 \text{ day}}{\text{shutdown}}\right) = \frac{0.69 \text{ day}}{\text{plant-yr}}$$

$$\text{BWR:} \quad \left(\frac{5.20 \text{ transients}}{\text{plant-yr}}\right)(0.80)(0.25)\left(\frac{0.75 \text{ day}}{\text{shutdown}}\right) = \frac{0.78 \text{ day}}{\text{plant-yr}}$$

Frequency Estimate

The parameters which included direct operator action were adjusted based on the 2% operator error reduction. In addition, the reduced transient frequency calculated from above were divided by the total PWR and BWR transient frequencies (i.e., 9.8 events/plant-yr for PWRs and 8.9 events/plant-yr for BWRs) to give a percent transient reduction. Then the parameters for transients (T_2 and T_3 for PWRs and T_{23} for BWRs) were adjusted.

Combining the reduction in operator error and the reduction in transient frequencies, the reductions in core-melt frequencies are 4.4×10^{-6} event/plant-yr for PWRs and 2.6×10^{-6} event/plant-yr for BWRs.

Consequence Estimate

The associated per-plant reduction in public risk was calculated (assuming 340 people per square mile) to be 12 man-rem/plant-yr for PWRs and 18 man-rem/plant-yr for BWRs. Assuming 90 PWRs and 44 BWRs with remaining lives of 28.8 and 27.4 years, respectively, the total public risk reduction was calculated to be 53,000 man-rem.

Cost Estimate

Industry Cost: For the advanced diagnostic system, implementation costs (hardware and installation), were estimated to be \$1.5M/plant. The on-line surveillance system was estimated to cost \$125,000/plant for hardware and \$375,000/plant for installation. For 134 plants, the total implementation cost is approximately \$270M.

Industry labor for operation and maintenance was estimated to be about 10 man-weeks/plant-year beyond that currently required for control room instrumentation. Therefore, this cost would be:

(10 man-wk/plant-yr)(\$2,270/man-wk)(134 plants)(30 years) = \$91M.

Therefore, the total industry cost was estimated to be \$360M.

NRC Cost: NRC costs for issue resolution were considered to be relatively minor (\$2M), based on the assumption that EPRI would continue to do the major portion of the research on this issue. NRC costs for labor to approve and monitor hardware changes to backfit plants were based on an average of 4 man-wk/backfit per plant. This cost is given by:

(4 man-wk/backfit plant)(\$2,270/man-wk)(71 plants) = \$650,000.

Therefore, the total NRC cost is \$2.65M.

Value/Impact Assessment

Based on a total public risk reduction of 53,000 man-rem, the value/impact score is given by:

$$S = \frac{53,000 \text{ man-rem}}{\$(360 + 2.65)M}$$

$$\cong 150 \text{ man-rem/\$M}$$

Uncertainty

The assumed benefits of resolution and cost for implementation of this safety issue are extremely hard to quantify because of the uncertain nature of possible future developments in this area.

Other Considerations

(1) If it is assumed that replacement power costs $300,000/day and, as previously calculated, the issue resolution will reduce down time by 0.69 day/plant-yr for PWRs and 0.78 day/plant-yr for BWRs, the industry cost saving is:

($300,000/day)[(0.69 day/plant-yr)(90 plants)(30 years) + (0.78 day/plant-yr) (44 plants)(30 yrs)] = $870M

Combining this with the industry costs (implementation and operation) would show an industry saving of about $500M. Including accident avoidance costs would further increase this saving.

(2) EPRI is doing research in this area which is being followed by NRC.

CONCLUSION

The calculated value/impact score barely indicated a medium priority, but the potential saving in plant downtime would make the implementation of the solution to this issue much more cost-effective. Based on these factors and the additional factor that required NRC resources were minimal, this issue was given a medium priority ranking. However, with the publication of NUREG-0985, Revision 1,[651] this item is now covered in Section 5.2 of the HFPP.

ITEM I.D.6: TECHNOLOGY TRANSFER CONFERENCE

DESCRIPTION

NRC and IEEE jointly sponsored a technology transfer conference in January, 1980. The conference was entitled "Advanced Electrotechnology Applications to Nuclear Power Plants," and had as its objective to consider the practicality of applying advanced technologies from other industries (e.g. aerospace, defense, aviation) to the nuclear power industry.

During the conference, eight parallel workshops were held including: Systems Management Techniques; Reliability Engineering; Risk Assessment; Software Reliability Verification and Validation; Smart Instrumentation; Operational

Aids-Command Control and Communications; Education, Training and Simulators; and Simulation and Analysis. The conference report[306] was issued in June 1980. This item is related to increasing knowledge and understanding of safety issues and, therefore, is considered a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

REFERENCES

48. NUREG-0660, "NRC Action Plan Item Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980.

55. Regulatory Guide 1.97 "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," U.S. Nuclear Regulatory Commission.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

98. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980.

150. Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," U.S. Nuclear Regulatory Commission, May 1973.

151. SECY-82-111, "Requirements for Emergency Response Capability," March 11, 1982.

152. NUREG/CR-2417, "Identification and Analysis of Human Errors Underlying Pump and Valve Related Events Reported by Nuclear Power Plant Licensees," U.S. Nuclear Regulatory Commission, February 1982.

153. "Safety System Status Monitoring: Draft Report on Current Industry Practice," Battelle Pacific Northwest Laboratories, June 1982.

241. NUREG/CR-1440, "Light Water Reactor Status Monitoring During Accident Conditions," U.S. Nuclear Regulatory Commission, May 1980.

242. NUREG/CR-2100, "Boiling Water Reactor Status Monitoring During Accident Conditions," U.S. Nuclear Regulatory Commission, May 1981.

243. NUREG/CR-2278, "Light Water Reactor Engineered Safety Features Status Monitoring," U.S. Nuclear Regulatory Commission, October 1981.

244. NUREG/CR-2147, "Nuclear Control Room Annunciators," U.S. Nuclear Regulatory Commission, October 1981.

245. RIL-124, "Control Room Alarms and Annunciators," U.S. Nuclear Regulatory Commission.

246. RIL-98, "Light Water Reactor Status Monitoring During Accident Conditions," U.S. Nuclear Regulatory Commission, August 18, 1980.

306. IEEE Catalog No. TH0073-7, "Record of the Working Conference on Advanced Electrotechnology Applications to Nuclear Power Plants, January 15-17, 1980, Washington, D.C." Institute of Electrical and Electronics Engineers.

307. EPRI NP-2230, "ATWS: A Reappraisal, Part 3," Electric Power Research Institute, 1982.

376. NRC Letter to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, "Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability (Generic Letter No. 82-33)," December 17, 1982.

474. NUREG-0700, "Guidelines for Control Room Design Reviews," U.S. Nuclear Regulatory Commission, September 1981.

651. NUREG-0985, Revision 1, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," U.S. Nuclear Regulatory Commission, September 1984.

TASK I.G: PREOPERATIONAL AND LOW-POWER TESTING

The objectives of this task are as follows: (1) to increase the capability of the shift crews to operate facilities in a safe and competent manner by assuring that training for plant changes and off-normal events is conducted. Near-term operating license facilities will be required to develop and implement intensified training exercises during the low-power testing programs; and (2) to review the comprehensiveness of test programs.

ITEM I.G.2: SCOPE OF TEST PROGRAM

DESCRIPTION

Historical Background

The major thrust of TMI Action Plan[48] Task I.G was to use the preoperational and startup test programs as a training exercise for the operating crews. In contrast to this, Subtask I.G.2 called for a more comprehensive test program to search for anomalies in the plant's response to a transient. This issue was suggested independently by the Kemeny Commission,[175] the Rogovin Commission,[161] the ACRS,[176] and the TMI Operations Team.[177]

Safety Significance

The safety significance of this issue lies in the early discovery of anomalies or unanticipated plant behavior. The TMI-2 accident is the most well-known example, but other less severe examples, such as the core-annulus water level decoupling at Oyster Creek, have taken place.

When the plant responds to a transient in an anomalous or unanticipated manner, the result may be an accident caused directly by the new phenomena, or an accident may result because of surprise or confusion on the part of the operators. The latter is probably the more likely of the two.

Possible Solution

The nature of the solution to this issue is implicit in its definition--an augmented test program. However, relatively little has been written concerning the nature and extent of this program. NUREG-0660[48] merely calls for the NRC to develop a program. Recommendations[177] made by an OIE team investigating TMI-2 are more specific: detailed review of all unscheduled transients during the first year as well as review of the preoperational and startup tests.

In actual fact, there is a spectrum of possible test programs ranging from the current program to programs which would take years. Morever, it may well not be necessary for each plant to perform each test. In addition, there is a large amount of data from operating experience which could supply information.

PRIORITY DETERMINATION

Frequency Estimate

Transients occur approximately ten times per reactor-year. However, most of these are relatively routine (e.g., turbine trip) and are thus unlikely to produce unpleasant surprises. In any case, present startup programs should cover them adequately. Therefore, we will focus our attention on transients which are rare, but are nevertheless frequent enough to be considered "anticipated operational occurrences." EPRI NP-801[178] is a report of the transients actually experienced in operating history. Based on judgment, we have selected transients which are candidates for suspicion of anomalous behavior.

PWR Transients	Frequency (RY^{-1})
Hi/Lo Pressurizer Pressure	0.10
Pressurizer Safety or Relief-Valve Opening	0.02
Inadvertent SIS	0.04
Loss of RCS Flow	0.04
Close All MSIVs	0.05
Sudden Opening of Secondary Relief Valves	0.06
Loss of Component Cooling	0.01
Loss of Service Water System	0.01
Total:	0.33

BWR Transients	Frequency (RY^{-1})
Pressure Regulator Fails Open	0.29
Pressure Regulator Fails Closed	0.14
Inadvertent Opening of S/RV	0.20
Trip One Recirculation Pump	0.02
Trip All Recirculation Pumps	0.06
Total:	0.71

Currently, reactor experience totals about 225 BWR-years and 340 PWR-years (565 RY total).[179] Thus, it is estimated that around 270 of the listed transients have occurred. Some of these transients have indeed illustrated the need for corrective measures. Unfortunately, it is not practical to use the computerized data banks to search for "anomalous behavior." Once again we are compelled to use judgment. At least four transients with anomalous response have occurred (Davis-Besse, Three Mile Island, Oyster Creek, Pilgrim) and are widely known. If a more thorough review of operating experience were made, more would be discovered. We estimate that perhaps 10 transients have shown some sort of unanticipated phenomenon. However, the number of interest is the number of phenomena left to be discovered. With about 270 transients of interest already history, anomalous events are not expected to be very common. Moreover, those discoveries which have been made have also led to measures intended to prevent future problems.

Bearing all this in mind, we estimate that anomalous or unanticipated behavior can be expected at a rate of about 5 events in 565 RY (i.e., half the estimated historical rate) or about 10^{-2}/RY. This number is an "educated guess" that the actual number of events that have occurred is higher than the four events listed but will be lower in the future because this experience has been used to correct these problems.

Consequence Estimate

Most anomalous transients have no consequences in the sense of releasing radioactivity. Based on the experience of TMI (one event in perhaps 10), we will assume that one event in 10 will result in core damage (extensive cladding failure) and one event in 100 will result in a core-melt with a significant release. We will approximate the former with a PWR-9 or BWR-5 and the latter with a PWR-7 or BWR-4.

We will assume that an augmented startup program will be 50% effective in discovering and correcting problems. The total risk reduction associated with this issue is 2.58×10^4 man-rem, based on 252 man-rem for 36 PWRs and 2.56×10^4 man-rem for 21 BWRs.

Cost Estimate

Industry Cost: As was stated previously, there is a spectrum of possible test programs. We will assume that the test program will average out to 2 weeks per plant. At $300,000/day for replacement power (which will dominate the cost), this is $4.2M per plant. The 2-week average estimate assumes that not every plant will perform every test. In many cases, the first of a given product line will perform a great deal of testing which will apply to all plants of the same design; or, testing could be shared within a product line by some other plan. Therefore, the total industry cost is $239.4M.

NRC Cost: For NRC cost, we will assume 5 staff-years to develop guidelines and approve generic plans, plus one staff-month of post-test review per plant. With 57 OLs on the docket (36 PWRs and 21 BWRs), this works out to about $1M.

Value/Impact Assessment

Based on a total risk reduction of 2.58×10^4 man-rem, the value/impact score is given by:

$$S = \frac{2.58 \times 10^4 \text{ man-rem}}{\$(239.4 + 1)\text{M}}$$

$$= 108 \text{ man-rem/\$M}$$

Uncertainties

The frequency estimates used here do not rest upon firm bases. This is not surprising since, like any other program where the goal is discovery, if good bases were available for estimates of effectiveness, the tests would not be necessary. Nevertheless, we can attempt to put bounds on our figures. The frequency of core damage is not likely to be uncertain to more than a factor of 10. If the true frequency were a factor of 10 higher, about 6 core-damaging accidents should have occurred by now. If it were a factor of 10 lower, the TMI-2 accident would have probability on the order of 5%.

However, the frequency of core-melt is subject to more uncertainty. We have assumed that the frequency of core-melt is one-tenth of that of core-damage. We will assume that this figure could be either a factor of 5 higher (every second TMI-like event a core-melt) or a factor of 5 lower (one core-melt in 50 core-damage events).

If we assume that the public dose estimates are uncertain to a factor of 5 and the costs to a factor of 5, then S would have a range from 3×10^0 to 4×10^3 man-rem/$M.

Other Considerations

The value/impact score obtained above does not consider the averted costs of cleanup. If such costs ($0.25M/RY) were included, the value/impact score would be significantly higher but not enough to justify a higher priority.

CONCLUSION

Based on the consideration of the value/impact score and the associated public risk, this item was determined to be a medium priority issue. However, with revisions to SRP[11] Section 14 and the OIE Manual, this issue was RESOLVED and new requirements were established.[654]

REFERENCES

11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980.

161. NUREG/CR-1250, "Three Mile Island: A Report to the Commission and to the Public," U.S. Nuclear Regulatory Commission, January 1980.

175. ZAR-791030-01, "Report of the President's Commission on the Accident at Three Mile Island," J. Kemeny, et. al., November 30, 1979.

176. Memorandum for J. Ahearne from M. Carbon, "Comments on the Pause in Licensing," December 11, 1979.

177. Memorandum for N. Moseley from J. Allan, "Operations Team Recommendations-IE/TMI Unit 2 Investigation," pp. 9, 36, October 16, 1979.

178. EPRI NP-801, "ATWS: A Reappraisal, Part III, Frequency of Anticipated Transients," Electric Power Research Institute, July 1978.

179. NUREG-0020, "Licensed Operating Reactors, Status Summary Report," U.S. Nuclear Regulatory Commission, February 1982.

180. NUREG-0580, "Regulatory Licensing Status Summary Report," U.S. Nuclear Regulatory Commission, June 1982.

654. Memorandum for W. Dircks from H. Thompson, "Closeout of TMI Action Plan Task I.G.2, 'Scope of Test Program,'" October 5, 1984.

TASK II.A: SITING

The objective of this task is to provide an added contribution to safety through the development of siting criteria for new power plants and the reevaluation with regard to the new siting criteria of facilities either under construction or operating.

ITEM II.A.1: SITING POLICY REFORMULATION

DESCRIPTION

In this TMI Action Plan item,[48] the staff was required to identify the principal criteria for evaluating proposed sites for nuclear power stations, recommend the adoption of these criteria in a Proposed Rule on Siting, and prepare an environmental assessment or environmental impact statement of the proposed revisions to meet NEPA requirements.

PRIORITY DETERMINATION

This issue was investigated by PNL but no risk or cost analyses were made.[64]

Frequency Estimate

Siting does not directly affect the frequency of radioactive releases. However, it should be noted that longer transmission lines will increase the frequency of load rejections and thus somewhat increase the probability of a release.

Consequence Estimate

For this case, we will use the WASH-1400[16] estimates of risk and assume a uniform population density of 340 people per square mile, which is the mean average for U.S. sites. If we multiply frequency by consequences for each release category and then sum the products, the average risks are 70 man-rem per PWR-year and 150 man-rem per BWR-year. Thus, compared to an average site, the maximum difference remote siting could make would be 150 man-rem/RY, which corresponds to locating a BWR in a completely deserted area. This average population density can be compared to the current criteria in SRP[11] Section 2.1.3 which limit the surrounding population density to about 500 people per square mile.

Cost Estimate

Industry Cost: Remote siting involves a number of cost factors. The following are considered most significant: transmission line losses; lower plant availability, due to longer transmission lines; cost of land for a major transmission line corridor, and delays involved in acquiring the land; and recruiting and relocating personnel to staff the plant.

The latter two, although widely recognized as significant, are difficult to quantify generically. However, if we assume a 1% line loss (reasonable for a 100-mile line) and five additional load rejections per year, the first two factors above total more than \$100M for a 1,000 MWe plant over 40 years.

NRC Cost: NRC costs are insignificant in comparison to industry costs.

Value/Impact Assessment

We can now estimate an upper limit on the value/impact score, based on a public risk reduction of 6,000 man-rem/reactor and a total cost of \$100M/reactor over a 40-year life.

$$S \leq \frac{6{,}000 \text{ man-rem/reactor}}{\$100\text{M/reactor}}$$

$$\leq 60 \text{ man-rem/\$M}$$

Other Considerations

The relatively low value/impact score must be combined with consideration of the net risks of 70 man-rem/PWR-year and 150 man-rem/BWR-year. Over a 40-year plant life, this corresponds to 3,000 to 6,000 man-rem, which would normally place this issue automatically in the high priority category, regardless of value/impact score or cost-effectiveness. However, this is the maximum risk reduction and most future sites would provide less; but specific sites may have better access to the grid and thus may be more cost-effective. However, there are no new plants being proposed.

CONCLUSION

Based on the above considerations and the need to address siting questions, this issue was given a medium priority. However, in 1984, the Commission decided to better define its safety objectives and better characterize radioactive source terms before proceeding with new siting regulations. As a result, it was decided that, before new siting efforts can be undertaken, a new radioactive source term must be approved and the evaluation of the safety goal must be completed. Upon completion of these two tasks, the need for a revised siting rule will be reassessed and, if necessary, a new generic safety issue will be established to address siting rulemaking. Thus, this item was RESOLVED and no new requirements were established.[655]

ITEM II.A.2: SITE EVALUATION OF EXISTING FACILITIES

DESCRIPTION

In this TMI Action Plan item,[48] the staff was to "...prepare an analysis for Commission decision of the NRC staff plans to reconsider, with regard to the revised siting policy, facilities either under construction or operating. The analysis would take, as a point of departure, the criteria expressed in the Proposed Rule on Siting (Item II.A.1) and would address a strategy for consideration of siting decisions of plants that have construction permits or operating licenses."

CONCLUSION

This issue was investigated by PNL.[64] The basic purpose behind this issue is now being addressed in the larger context of the Safety Goal[69] which is being developed under TMI Action Plan[48] Item V.A.1. Consequently, all NRC staff efforts on this issue were terminated in mid-1981.

REFERENCES

16. WASH-1400 (NUREG-75/014), "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, October 1975.

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

69. NUREG-0880, "Safety Goals for Nuclear Power Plants: A Discussion Paper," U.S. Nuclear Regulatory Commission, February 1982.

655. Memorandum for W. Dircks from H. Denton, "Generic Issue II.A.1, 'Siting Policy Reformulation,'" September 17, 1984.

TASK II.E.5: DESIGN SENSITIVITY OF B&W REACTORS

The objective of this task is to reduce the sensitivity of B&W plants to feedwater transients, with emphasis on the overcooling transients that have been observed at B&W operating plants.

ITEM II.E.5.1: DESIGN EVALUATION

DESCRIPTION

Historical Background

The NRC staff concluded that B&W reactors exhibited unique sensitivity to secondary system transients (both undercooling and overcooling events). Therefore, B&W plants under construction were required to propose recommendations on hardware and procedure changes relative to the need for methods for damping primary system sensitivity to perturbations in the once-through steam generator (OTSG). This issue also considered the backfitting of the recommendations on operating plants.

Safety Significance

The safety significance of this TMI Action Plan item[48] is the same as that for Item II.E.5.2 i.e., the perception of what constitutes acceptable response to transients.

Possible Solution

All B&W plants under construction were required [10 CFR 50.54(f)] to provide recommendations to reduce plant sensitivity.[154] The recommendations (with proposed modifications) were submitted for NRC review.

The staff also evaluated the modifications proposed by the applicants for possible backfit to operating plants.[159,160,547] The staff concluded that the portion of this issue which deals with plants under construction was completed with the issuance of the Midland 1&2 SER which evaluated the modifications.[159,160,547] The other B&W plants under construction will be evaluated as part of the normal licensing review.

The portion of the issue which deals with backfit considerations was also completed.[159,160,547] Specifically, the staff concluded that the Midland modifications would be effective in reducing both the frequency and severity of overcooling transients and recommended that similar modifications be made at operating B&W plants. They also concluded that there are a number of related activities underway. These are as follows:

1. Operating B&W plants are implementing upgrades to meet NUREG-0737.[98]

2. USI A-47, "Safety Implications of Control Systems," is evaluating steam generator overcooling/overfilling as it relates to control system failures.

3. The staff is also pursuing resolution of overcooling events (steam bubble formation/natural circulation interruption) on a generic basis with the B&W Owners' Group [NUREG-0737,[98] Item II.K.3(30)].

4. Consideration of pressurized thermal shock (PTS) concerns relating to overcooling are being addressed by the staff as part of the resolution of USI A-49, "Pressurized Thermal Shock."

Based on the above, the staff concluded that the B&W-designed operating reactors have responded to staff concerns regarding the frequency of overcooling and steam generator overfill events by implementing plant modifications. The adequacy of these modifications will be confirmed by other ongoing programs.

CONCLUSION

This item was RESOLVED and requirements were established.

ITEM II.E.5.2: B&W REACTOR TRANSIENT RESPONSE TASK FORCE

DESCRIPTION

Historical Background

After TMI-2, the NRC staff investigated[155] the response of B&W reactors to transients and determined that, in their opinion, they are overly responsive to certain transients. This responsiveness or sensitivity was attributed to a number of design and operational features including the small secondary water inventory in the steam generator, the small pressurizer volume, the pilot-operated relief valve (PORV) set-point, and the high pressure injection (HPI) set-points. As a result of the investigation, a number of recommendations were made for improving the plant response.[155]

The recommendations covered a number of design changes and operational considerations. DST provided a prioritization for the recommendations[158] in August of 1980. A number of these recommendations (referred to as Category A items) have already been implemented (or are being implemented) for the B&W operating plants.[156,157] The other recommendations (referred to as Category B items) have not been issued as requirements although a number of them have been implemented by some utilities with B&W plants as part of their own investigations.

Safety Significance

The safety significance of this TMI Action Plan[48] item depends on the perception of what constitutes acceptable response to transients. Present NRC requirements are outlined in the SRP[11] and all plants are required to meet these as a minimum. It has been suggested by DSI[159] that additional performance criteria are necessary to more restrict the plants' response to transients and as a result limit the potential for plant damage.

Possible Solution

The technical resolution of this issue was defined in NUREG-0667.[155] It was suggested[159] that to implement the resolution required additional specification of the staff's performance criteria for transient response. (Present criteria are contained in the SRP.[11]) Therefore, DSI proposed[159] that a uniform requirement in the form of criteria should be issued by NRC to ensure that adequate steps are taken by all B&W plants. Specifically, the recommended criteria are:

1. ECCS actuation or loss of pressurizer level indication should not normally occur following a reactor trip or main feed water control failure.

2. Credit for operator action to mitigate overcooling events should be consistent with the guidelines of ANSI N660.[45]

3. Steam generators should be protected from overfill from main or auxiliary feedwater flow to limit overcooling. This equipment should be safety grade if flooding of the steam lines is an unanalyzed event.

Based on a DST evaluation[160] of this issue, it was recommended that implementation would be best accomplished by issuance of a statement of NRC's performance criteria for transients. It was also recommended that the first two criteria and accompanying value/impact statements be submitted to CRGR for review. The third criterion was included in USIs A-47 and A-49.

CONCLUSION

This issue was RESOLVED and requirements were established.[656,657]

REFERENCES

11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.

45. ANSI N660, "Time Response Design Criteria for Safety-Related Operator Actions," American National Standards Institute.

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980.

98. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1983.

154. NRC Letter to Construction Permit Holders of B&W Designed Facilities, October 25, 1979.

155. NUREG-0667, "Transient Response of Babcock & Wilcox Designed Reactors," U.S. Nuclear Regulatory Commission, May 1980.

156. Memorandum for H. Denton from D. Eisenhut, "NUREG-0667, Transient Response of Babcock & Wilcox Designed Reactors, Implementation Plan," June 3, 1981.

157. Memorandum for D. Eisenhut from G. Lainas, "Status Report on Implementation of NUREG-0667 Category A Recommendations," December 15, 1981.

158. Memorandum for H. Denton from R. Mattson, "Review of Final Report of the B&W Reactor Transient Response Task Force (NUREG-0667)," August 8, 1980.

159. Memorandum for S. Hanauer from R. Mattson, "Design Sensitivity of B&W Reactors, Item II.E.5.1 of NUREG-0660," February 26, 1982.

160. Memorandum for R. Mattson from S. Hanauer, "Design Sensitivity of B&W Reactors," June 21, 1982.

547. Memorandum for W. Dircks from R. Mattson, "Closeout of NUREG-0660 Item II.E.5.1 Design Sensitivity of B&W Plants for Operating Plants," March 15, 1983.

656. Memorandum for W. Dircks from H. Denton, "Closeout of TMI Action Plan Task II.E.5.2, Transient Response of B&W Designed Reactors," September 23, 1984.

657. Memorandum for D. Crutchfield from D. Eisenhut, "TMI Action Plan Task II.E.5.2," November 6, 1984.

TASK II.K: MEASURES TO MITIGATE SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS AND LOSS-OF-FEEDWATER ACCIDENTS

The objectives of this task were to perform systems reliability analyses and to effect changes in emergency operating procedures and operator training to improve the capability of plants to mitigate the consequences of the small-break LOCAs and loss-of-feedwater events.

ITEM II.K.1: IE BULLETINS

Between April 1, 1979 and July 26, 1979, OIE issued 9 bulletins to various operating plants, depending on their reactor design, and a review of the affected licensee responses was conducted by the NRR Bulletins and Orders Task Force (BOTF). The responses were determined to be acceptable and separate evaluation reports were prepared and issued to some licensees. Thus, prior to the publication of NUREG-0660,[48] several parts of this item were either completed or found to be covered in other TMI Action Plan items. This status was reported in Table C.1 of NUREG-0660.[48] The following is a summary of the 28 parts of this item.

ITEM II.K.1(1): REVIEW TMI-2 PNs AND DETAILED CHRONOLOGY OF THE TMI-2 ACCIDENT

DESCRIPTION

This NUREG-0660[48] item affected all OLs and was originated to effect short-term changes in emergency operating procedures and operator training in order to improve the capability of plants to mitigate the likelihood and consequences of SBLOCAs and loss-of-feedwater events. For all OL applicants, this item was determined to be covered by Items I.A.2.2 and I.A.3.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(2): REVIEW TRANSIENTS SIMILAR TO TMI-2 THAT HAVE OCCURRED AT OTHER FACILITIES AND NRC EVALUATION OF DAVIS-BESSE EVENT

DESCRIPTION

This NUREG-0660[48] item affected all B&W operating plants. For OL applicants with B&W reactors, this item was determined to be covered by Items I.A.2.2 and I.A.3.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(3): REVIEW OPERATING PROCEDURES FOR RECOGNIZING, PREVENTING, AND MITIGATING VOID FORMATION IN TRANSIENTS AND ACCIDENTS

DESCRIPTION

This NUREG-0660[48] item affected all operating PWRs. For OL applicants with PWRs, it was determined that the issue was covered by Item I.C.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(4): REVIEW OPERATING PROCEDURES AND TRAINING INSTRUCTIONS

DESCRIPTION

This NUREG-0660[48] item was divided into 4 parts to ensure: (a) that operators do not override ESF actions unless continued operation is unsafe; (b) HPI system operation; (c) RCP operation; and (d) that operators are instructed not to rely on level indication alone in evaluating plant conditions.

- Part (a) affected all operating plants. However, for all OL applicants it was determined that this part was covered by Items I.C.1, I.C.7, I.G.1, and I.C.8.

- Part (b) affected all W, CE and B&W operating plants with specific requirements issued to ANO-1; Davis-Besse 1; Oconee 1, 2, and 3; Crystal River 3; and Rancho Seco. For OL applicants with W, CE, or B&W reactors, it was determined that this part was covered by Item I.C.1.

- Part (c) affected all PWRs and was completed by OLs prior to the publication of NUREG-0660.[48] For OL applicants with PWRs, it was determined that this part was covered by Items I.C.1 and I.A.1.3.

- Part (d) affected all plants and was completed by OLs prior to the publication of NUREG-0660. For all OL applicants, it was determined that this part was covered by Items I.C.1, I.A.3.1, and II.F.2.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(5): SAFETY-RELATED VALVE POSITION DESCRIPTION

DESCRIPTION

This NUREG-0660[48] item was divided into 2 parts and required plants to: (a) review all valve positions and positioning requirements and positive controls along with all related test and and maintenance procedures to assure proper ESF functioning, if required; and (b) verify that AFW valves are in the open position.

- Part (a) affected all operating plants. For all OL applicants, it was determined that this part was covered by Items I.C.2 and I.C.6.

- Part (b) affected all B&W operating plants. For OL applicants with B&W reactors, this part was also determined to be covered by Items I.C.2 and I.C.6.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(6): REVIEW CONTAINMENT ISOLATION INITIATION DESIGN AND PROCEDURES

DESCRIPTION

This NUREG-0660[48] item affected all operating plants and was initiated to assure isolation of all lines that do not degrade safety features or cooling capability upon automatic initiation of SI. For all OL applicants, it was determined that this issue was covered by Item II.E.4.2.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(7): IMPLEMENT POSITIVE POSITION CONTROLS ON VALVES THAT COULD COMPROMISE OR DEFEAT AFW FLOW

DESCRIPTION

This NUREG-0660[48] item affected all B&W operating plants. For OL applicants with B&W reactors, this issue was determined to be covered by Item II.E.1.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(8): IMPLEMENT PROCEDURES THAT ASSURE TWO INDEPENDENT 100% AFW FLOW PATHS

DESCRIPTION

This NUREG-0660[48] item required all operating B&W plants to immediately implement procedures that assure two independent 100% AFW flow paths or specify explicitly LCO with reduced AFW capacity. For OL applicants with B&W reactors, this issue was determined to be covered by Item II.E.1.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(9): REVIEW PROCEDURES TO ASSURE THAT RADIOACTIVE LIQUIDS AND GASES ARE NOT TRANSFERRED OUT OF CONTAINMENT INADVERTENTLY

DESCRIPTION

This NUREG-0660[48] item required all operating plants to review their procedures to assure that radioactive liquids and gases are not transferred out of containment inadvertently, especially upon ESF reset. All applicable systems and interlocks were required to be listed. For OL applicants, this item was determined to be covered by Items II.E.4.2 and I.C.6.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(10): REVIEW AND MODIFY PROCEDURES FOR REMOVING SAFETY-RELATED SYSTEMS FROM SERVICE

DESCRIPTION

This NUREG-0660[48] item required all operating plants to review and modify (as required) their procedures for removing safety-related systems from service (and restoring to service) to assure operability status is known. For OL applicants, the issue was determined to be covered by Items I.C.2 and I.C.6.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(11): MAKE ALL OPERATING AND MAINTENANCE PERSONNEL AWARE OF THE SERIOUSNESS AND CONSEQUENCES OF THE ERRONEOUS ACTIONS LEADING UP TO, AND IN EARLY PHASES OF, THE TMI-2 ACCIDENT

DESCRIPTION

This NUREG-0660[48] item affected all operating plants. For OL applicants, the issue was determined to be covered by Items I.A.2.2 and I.A.3.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(12): ONE HOUR NOTIFICATION REQUIREMENT AND CONTINUOUS COMMUNICATIONS CHANNELS

DESCRIPTION

This NUREG-0660[48] item affected all operating plants. For OL applicants, the issue was determined to be covered by Items I.E.6 and III.A.3.3.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(13): PROPOSE TECHNICAL SPECIFICATION CHANGES REFLECTING IMPLEMENTATION OF ALL BULLETIN ITEMS

DESCRIPTION

This NUREG-0660[48] item required all operating plants to propose TS changes reflecting implementation of all Bulletin items, as required.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(14): REVIEW OPERATING MODES AND PROCEDURES TO DEAL WITH SIGNIFICANT AMOUNTS OF HYDROGEN

DESCRIPTION

This NUREG-0660[48] item affected all operating plants with W, CE, and GE reactors. For OL applicants with W, CE and GE reactors, it was determined that the issue was covered by Items II.B.4, II.B.7, II.E.4.1, and II.F.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(15): FOR FACILITIES WITH NON-AUTOMATIC AFW INITIATION, PROVIDE DEDICATED OPERATOR IN CONTINUOUS COMMUNICATION WITH CR TO OPERATE AFW

DESCRIPTION

This NUREG-0660[48] item affected all operating plants with W and CE reactors. However, prior to the publication of NUREG-0660,[48] all necessary action was completed by the affected OLs. For OL applicants with W and CE reactors, it was determined that the issue was covered by Item II.E.1.2.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(16): IMPLEMENT PROCEDURES THAT IDENTIFY PRZ PORV "OPEN" INDICATIONS AND THAT DIRECT OPERATOR TO CLOSE MANUALLY AT "RESET" SETPOINT

DESCRIPTION

This NUREG-0660[48] item affected all operating plants with W and CE reactors. However, prior to the publication of NUREG-0660,[48] all necessary action was completed by the affected OLs. For OL applicants with W and CE reactors, it was determined that the issue was covered by Items I.C.1 and II.D.3.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(17): TRIP PZR LEVEL BISTABLE SO THAT PZR LOW PRESSURE WILL INITIATE SAFETY INJECTION

DESCRIPTION

This NUREG-0660[48] item required all OLs and OL applicants with W reactors to trip the pressurizer level bistable so that the pressurizer low pressure (rather than the pressurizer low pressure and pressurizer low level coincidence) would initiate safety injection. For testing, the plants were required to reset the low level bistable. However, prior to the publication of NUREG-0660,[48] all necessary action was completed by the affected OLs.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(18): DEVELOP PROCEDURES AND TRAIN OPERATORS ON METHODS OF ESTABLISHING AND MAINTAINING NATURAL CIRCULATION

DESCRIPTION

This NUREG-0660[48] item affected all operating B&W plants. However, prior to the publication of NUREG-0660, all necessary action was completed by the affected plants. For OL applicants with B&W reactors, it was determined that the issue was covered by Items I.C.1 and I.G.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(19): DESCRIBE DESIGN AND PROCEDURE MODIFICATIONS TO REDUCE LIKELIHOOD OF AUTOMATIC PZR PORV ACTUATION IN TRANSIENTS

DESCRIPTION

This NUREG-0660[48] item required all operating B&W plants to describe their design and procedure modifications (based on analysis) to reduce the likelihood of automatic pressurizer PORV actuation in transients. For OL applicants with B&W reactors, it was determined that the issue was covered by Item II.E.5.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(20): PROVIDE PROCEDURES AND TRAINING TO OPERATORS FOR PROMPT MANUAL REACTOR TRIP FOR LOFW, TT, MSIV CLOSURE, LOOP, LOSG LEVEL, AND LO PZR LEVEL

DESCRIPTION

This NUREG-0660[48] item affected all OLs and OL applicants with B&W reactors.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(21): PROVIDE AUTOMATIC SAFETY-GRADE ANTICIPATORY REACTOR TRIP FOR LOFW, TT, OR SIGNIFICANT DECREASE IN SG LEVEL

DESCRIPTION

This NUREG-0660[48] item affected all OLs and OL applicants with B&W reactors.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(22): DESCRIBE AUTOMATIC AND MANUAL ACTIONS FOR PROPER FUNCTIONING OF AUXILIARY HEAT REMOVAL SYSTEMS WHEN FW SYSTEM NOT OPERABLE

DESCRIPTION

This NUREG-0660[48] item affected all OLs and OL applicants with BWRs.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(23): DESCRIBE USES AND TYPES OF RV LEVEL INDICATION FOR AUTOMATIC AND MANUAL INITIATION SAFETY SYSTEMS

DESCRIPTION

This NUREG-0660[48] item required all OLs and OL applicants with BWRs to describe their uses and types of reactor vessel indication for automatic and manual initiation safety systems. The affected plants were also required to describe their alternative instrumentation.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(24): PERFORM LOCA ANALYSES FOR A RANGE OF SMALL-BREAK SIZES AND A RANGE OF TIME LAPSES BETWEEN REACTOR TRIP AND RCP TRIP

DESCRIPTION

This NUREG-0660[48] item affected all operating PWRs. However, prior to the publication of NUREG-0660,[48] all necessary action was completed by the affected OLs. For OL applicants with PWRs, the issue was determined to be covered by Item I.C.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(25): DEVELOP OPERATOR ACTION GUIDELINES

DESCRIPTION

This NUREG-0660[48] item required all operating PWRs to develop operator action guidelines, based on the analyses performed in response to Item II.K.1(24). However, prior to the publication of NUREG-0660,[48] it was determined that all necessary action was completed by the affected plants. For OL applicants with PWRs, the issue was determined to be covered by Item I.C.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(26): REVISE EMERGENCY PROCEDURES AND TRAIN ROs AND SROs

DESCRIPTION

This NUREG-0660[48] item required all operating PWRs to revise their emergency procedures and train ROs and SROs, based on guidelines developed in response to Item II.K.1(25). However, prior to the publication of NUREG-0660,[48] all necessary action was completed by the affected OLs. For OL applicants with PWRs, it was determined that the issue was covered by Items I.A.3.1, I.C.1, and I.G.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(27): PROVIDE ANALYSES AND DEVELOP GUIDELINES AND PROCEDURES FOR INADEQUATE CORE COOLING CONDITIONS

DESCRIPTION

This NUREG-0660[48] item required all operating PWRs to provide analyses and develop guidelines and procedures for inadequate core cooling conditions. The affected plants were also required to define their RCP restart criteria. However, prior to the publication of NUREG-0660,[48] all necessary action was completed by the affected OLs. For OL applicants with PWRs, it was determined that the issue was covered by Items I.C.1 and II.F.2.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.1(28): PROVIDE DESIGN THAT WILL ASSURE AUTOMATIC RCP TRIP FOR ALL CIRCUMSTANCES WHERE REQUIRED

DESCRIPTION

This NUREG-0660[48] item affected all operating PWRs. For OL applicants with PWRs,it was determined that the issue was covered by Item II.K.3(5).

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.2: COMMISSION ORDERS ON BABCOCK AND WILCOX PLANTS

This item contained 21 requirements for 7 operating plants with B&W reactors that were issued confirmatory shutdown orders shortly after the TMI-2 accident. Some of these requirements were also applicable to OL applicants with B&W reactors. These requirements were divided into two groups: short-term actions and long-term actions. The short-term actions were essentially those that were listed in Table 2-1 of NUREG-0645[580] while the long-term actions were also dilineated in NUREG-0645.[580]

However, prior to the publication of NUREG-0660,[48] 10 of these requirements were either completed or found to be covered by other TMI Action Plan items. This status was reported in Table C.2 of NUREG-0660.[48] Since that time, some of the remaining items have been clarified in NUREG-0737[98] and others have been completed. The status of the MPAs established for implementation can be found in NUREG-0748.[578] The following is a summary of the 21 parts of this item.

ITEM II.K.2(1): UPGRADE TIMELINESS AND RELIABILITY OF AFW SYSTEM

DESCRIPTION

All 7 B&W plants with OLs completed this short-term NUREG-0660[48] action before they were permitted to restart. These accomplishments were made in July 1979, prior to the publication of NUREG-0660.[48] For OL applicants with B&W reactors, it was determined that the issue was being addressed by Items II.E.1.1 and II.E.1.2.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.2(2): PROCEDURES AND TRAINING TO INITIATE AND CONTROL AFW INDEPENDENT OF INTEGRATED CONTROL SYSTEM

DESCRIPTION

All 7 B&W plants with OLs completed this short-term NUREG-0660[48] action before they were permitted to restart. These accomplishments were made prior to the publication of NUREG-0660.[48] This requirement was also applicable to OL applicants with B&W reactors and was clarified in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.2(3): HARD-WIRED CONTROL-GRADE ANTICIPATORY REACTOR TRIPS

DESCRIPTION

All 7 B&W plants with OLs completed this short-term NUREG-0660[48] action before they were permitted to restart. These accomplishments were made in July 1979, prior to the publication of NUREG-0660.[48] This requirement was not applicable to OL applicants with B&W reactors.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.2(4): SMALL-BREAK LOCA ANALYSIS, PROCEDURES AND OPERATOR TRAINING

DESCRIPTION

All 7 B&W plants with OLs completed this short-term NUREG-0660[48] action before they were permitted to restart. These accomplishments were made in September 1979, prior to the publication of NUREG-0660.[48] For OL applicants with B&W reactors, it was determined that the issue was being addressed by Items I.A.3.1 and I.C.1.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.2(5): COMPLETE TMI-2 SIMULATOR TRAINING FOR ALL OPERATORS

DESCRIPTION

All 7 B&W plants with OLs completed this short-term NUREG-0660[48] action before they were permitted to restart. These accomplishments were made prior to the publication of NUREG-0660.[48] For OL applicants with B&W reactors, it was determined that the issue was being addressed by Item I.A.2.6.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.2(6): REEVALUATE ANALYSIS FOR DUAL-LEVEL SETPOINT CONTROL

DESCRIPTION

Prior to the TMI-2 accident, Toledo Edison Company (TECO) was authorized by the NRC (pending incorporation of permanent design modifications to provide automatic dual setpoint steam generator level control) to manually control steam generator level at 35 in. for all events requiring auxiliary feedwater, unless a safety feature actuation system Level 2 signal occurred. Following the TMI-2 accident, the staff required additional information to verify that the effects of manually controlling steam generator level at 35 in. was adequate for the Davis-Besse plant, in light of the revised small-break LOCA analyses that were performed by B&W after the TMI-2 accident.

The only operating plant affected by this item, Davis-Besse 1, completed this short-term NUREG-0660[48] action before it was permitted to restart. This accomplishment was made in July 1979, prior to the publication of NUREG-0660.[48] This requirement was not applicable to OL applicants with B&W reactors.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.2(7): REEVALUATE TRANSIENT OF SEPTEMBER 24, 1977

DESCRIPTION

In September 1977, Davis-Besse 1 experienced an event which started out very similar to the one that occurred at TMI-2. In light of the information gained from the TMI-2 accident, the staff felt it was necessary to review the previous evaluation prepared by Toledo Edison Company for the Davis-Besse 1 event which involved equipment problems and depressurization of the primary system.

The only plant affected by this item, Davis-Besse 1, completed this short-term NUREG-0660[48] action before it was permitted to restart. This accomplishment was made in July 1979, prior to the publication of NUREG-0660.[48] This requirement was not applicable to OL applicants with B&W reactors.

CONCLUSION

The item was RESOLVED and requirements were issued.

ITEM II.K.2(8): CONTINUED UPGRADING OF AFW SYSTEM

DESCRIPTION

All 7 B&W plants with OLs were initially required to complete this long-term NUREG-0660[48] action. However, a clarification was issued in NUREG-0737[98] which superseded this item with Items II.E.1.1 and II.E.1.2. For OL applicants with B&W reactors, it was determined that the issue was being addressed by Items II.E.1.1 and II.E.1.2.

CONCLUSION

This item is covered in Items II.E.1.1 and II.E.1.2.

ITEM II.K.2(9): ANALYSIS AND UPGRADING OF INTEGRATED CONTROL SYSTEM

DESCRIPTION

This NUREG-0660[48] item called for licensees with B&W reactors to provide a failure mode effects analysis on the integrated control system. All 7 B&W plants with OLs as well as OL applicants with B&W reactors were required to complete this long-term action. A clarification that affected both groups of plants was issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-27 was established by DL for implementation purposes.

ITEM II.K.2(10): HARD-WIRED SAFETY-GRADE ANTICIPATORY REACTOR TRIPS

DESCRIPTION

This NUREG-0660[48] item called for licensees with B&W reactors to provide a design and schedule for implementation of a safety-grade reactor trip upon loss of feedwater, turbine trip, and significant reduction in steam generator level. These requirements were listed as Item 5 of IE Bulletin 79-05B which was issued on April 21, 1979. All 7 B&W plants with OLs as well as OL applicants with B&W reactors were required to complete this long-term action. Clarifications that affected both groups of plants were issued in NUREG-0737[98] and OL applicants with B&W reactors were given the option of complying with Item II.K.1 (Part C.1.21) of NUREG-0694[579] to satisfy this requirement.

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-28 was established by DL for implementation purposes.

ITEM II.K.2(11): OPERATOR TRAINING AND DRILLING

DESCRIPTION

All 7 B&W plants with OLs were required to complete this long-term NUREG-0660[48] item which called for continued operator training and drilling to assure a high state of preparedness. For the affected OLs, a clarification to the requirement was issued in NUREG-0737.[98] For OL applicants with B&W reactors, this item was determined to be covered Items I.A.2.2, I.A.2.5, I.A.3.1, and I.G.1 prior to the publication of NUREG-0660.[48]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-29 was established by DL for implementation purposes.

ITEM II.K.2(12): TRANSIENT ANALYSIS AND PROCEDURES FOR MANAGEMENT OF SMALL BREAKS

DESCRIPTION

The only operating B&W plant affected by this NUREG-0660[48] item was Davis-Besse 1. However, prior to the publication of NUREG-0660,[48] it was determined that the issue was covered by Item I.C.1 for Davis-Besse 1 and all OL applicants with B&W reactors.

CONCLUSION

This item is covered in Item I.C.1.

ITEM II.K.2(13): THERMAL-MECHANICAL REPORT ON EFFECT OF HPI ON VESSEL INTEGRITY FOR SMALL-BREAK LOCA WITH NO AFW

DESCRIPTION

This item required the affected plants to demonstrate that sufficient mixing of the high pressure injection water would occur with the reactor coolant so that significant thermal shock effects to the reactor vessel would be precluded. All 7 B&W plants with OLs and all OL applicants with B&W reactors were required to comply with this NUREG-0660[48] item. A clarification was issued in NUREG-0737[48] to include all PWRs (OLs and OL applicants).

CONCLUSION

This item was RESOLVED, requirements were issed, and MPA F-30 was established by DL for implementation purposes.

ITEM II.K.2(14): DEMONSTRATE THAT PREDICTED LIFT FREQUENCY OF PORVs AND SVs IS ACCEPTABLE

DESCRIPTION

All 7 B&W plants with OLs and all OL applicants with B&W reactors were required to comply with this NUREG-0660[48] item. A clarification affecting both groups of plants was issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-31 was established by DL for implementation purposes.

ITEM II.K.2(15): ANALYSIS OF EFFECTS OF SLUG FLOW ON ONCE-THROUGH STEAM GENERATOR TUBES AFTER PRIMARY SYSTEM VOIDING

DESCRIPTION

All 7 B&W plants with OLs and all OL applicants with B&W reactors were required to comply with this NUREG-0660[48] item which called for the affected plants to assess the loading on steam generator tube sheets induced from slug flow during natural circulation cooldown. A clarification affecting both groups of plants was issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.2(16): IMPACT OF RCP SEAL DAMAGE FOLLOWING SMALL-BREAK LOCA WITH LOSS OF OFFSITE POWER

DESCRIPTION

All 7 B&W plants with OLs and all OL applicants with B&W reactors were required to comply with this NUREG-0660[48] item which called for the investigation of the consequences of losing coolant to the seals of the reactor coolant pumps during loss of offsite power. Clarifications affecting both groups of plants were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-32 was established by DL for implementation purposes.

ITEM II.K.2(17): ANALYSIS OF POTENTIAL VOIDING IN RCS DURING ANTICIPATED TRANSIENTS

DESCRIPTION

All 7 B&W plants with OLs were required to comply with this NUREG-0660[48] item which called for the plants to determine the consequence of voiding in the reactor vessel and the hot legs during normal anticipated transients. For OL applicants with B&W reactors, it was determined that the issue was being addressed by Item I.C.1. Clarifications were issued in NUREG-0737[98] to include all PWRs (OLs and OL applicants).

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-33 was established by DL for implementation purposes.

ITEM II.K.2(18): ANALYSIS OF LOSS OF FEEDWATER AND OTHER ANTICIPATED TRANSIENTS

DESCRIPTION

All 7 B&W plants with OLs and all OL applicants with B&W reactors plants were affected by this NUREG-0660[48] item. However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Item I.C.1.

CONCLUSION

This item is covered in by Item I.C.1.

ITEM II.K.2(19): BENCHMARK ANALYSIS OF SEQUENTIAL AFW FLOW TO ONCE-THROUGH STEAM GENERATOR

DESCRIPTION

All 7 B&W plants with OLs were required to comply with this NUREG-0660[48] item which called for the evaluation of the steam generator model in the small-break licensing code (CRAFT-2) by predicting the Crystal River asymmetric cooldown start-up test. For OL applicants with B&W reactors, it was determined that the issue was being addressed by Item I.C.1. Clarifications were issued in NUREG-0737[98] to include all PWRs (OLs and OL applicants).

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-34 was established by DL for implementation purposes.

ITEM II.K.2(20): ANALYSIS OF STEAM RESPONSE TO SMALL-BREAK LOCA THAT CAUSES SYSTEM PRESSURE TO EXCEED PORV SETPOINT

DESCRIPTION

All 7 B&W plants with OLs were required to comply with this NUREG-0660[48] item which called for the assessment of small-break LOCAs which result in pressurization of the primary system to the PORV setpoint. For OL applicants with B&W reactors, it was determined that the issue was being addressed by Item I.C.1. A clarification affecting the 7 OLs was issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-35 was established by DL for implementation purposes.

ITEM II.K.2(21): LOFT 3-1 PREDICTIONS DESCRIPTION

DESCRIPTION

The adequacy of B&W's small-break LOCA model needed to be benchmarked against integral systems test data. By performing this pretest prediction of LOFT L3-1, the staff was able to determine this information. All 7 B&W plants were affected by this NUREG-0660[48] item which was completed in December 1979, prior to the publication of NUREG-0660.[48] OL applicants with B&W reactors were not affected by this item.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.3: FINAL RECOMMENDATIONS OF BULLETINS AND ORDERS TASK FORCE

This item contained 57 requirements that affected OLs and OL applicants. These requirements were based on recommendations that were developed by the staff and issued in the following reports: NUREG-0565[96] (B&W reactors), NUREG-0611[93] (W reactors), NUREG-0626[94] (GE reactors), NUREG-0635[95] (CE reactors), and NUREG-0623.[97] However, prior to the publication of NUREG-0660,[48] some of these requirements were superseded by other TMI Action Plan items. This status was reported in Table C.3 of NUREG-0660.[48] Since that time, some of the remaining items have been clarified in NUREG-0737[98] and others have been completed. The status of the MPAs established for implementation can be found in NUREG-0748.[578] The following is a summary of the 57 parts of this item.

ITEM II.K.3(1): INSTALL AUTOMATIC PORV ISOLATION SYSTEM AND PERFORM OPERATIONAL TEST

DESCRIPTION

This NUREG-0660[48] item required all operating PWRs to provide a system that uses the PORV block valve to protect against a small-break LOCA. This system will automatically cause the block valve to close when the reactor coolant system pressure decays after the PORV has opened. OL applicants with PWRs were also required to complete this item. Clarifications affecting both groups of plants were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-36 was established by DL for implementation purposes.

ITEM II.K.3(2): REPORT ON OVERALL SAFETY EFFECT OF PORV ISOLATION SYSTEM

DESCRIPTION

This NUREG-0660[48] item required all operating PWRs to document the action to be taken to decrease the probability of a small-break LOCA caused by a stuck-open

PORV. OL applicants with PWRs were also required to complete this item. Clarifications affecting both groups of plants were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-37 was established by DL for implementation purposes.

ITEM II.K.3(3): REPORT SAFETY AND RELIEF VALVE FAILURES PROMPTLY AND CHALLENGES ANNUALLY

DESCRIPTION

This NUREG-0660[48] item required all operating plants to report safety and relief valve failures promptly and challenges annually. All OL applicants were also required to complete this item. Clarifications affecting both groups of plants were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-38 was established by DL for implementation purposes.

ITEM II.K.3(4): REVIEW AND UPGRADE RELIABILITY AND REDUNDANCY OF NON-SAFETY EQUIPMENT FOR SMALL-BREAK LOCA MITIGATION

DESCRIPTION

This NUREG-0660[48] item only affected OL applicants. However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Items II.C.1, II.C.2, and II.C.3.

CONCLUSION

This item is covered in Items II.C.1, II.C.2, and II.C.3.

ITEM II.K.3(5): AUTOMATIC TRIP OF REACTOR COOLANT PUMPS

DESCRIPTION

This NUREG-0660[48] item required all PWR operating plants to study the need for automatic trip of RCPs and to modify procedures or designs, as appropriate. OL applicants with PWRs were also required to complete this item. Clarifications affecting both groups of plants were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-39 was established by DL for implementation purposes.

ITEM II.K.3(6): INSTRUMENTATION TO VERIFY NATURAL CIRCULATION

DESCRIPTION

This NUREG-0660[48] item affected all PWRs (OLs and OL applicants). However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Items I.C.1, II.F.2, and II.F.3.

CONCLUSION

This item is covered in Items I.C.1, II.F.2, and II.F.3.

ITEM II.K.3(7): EVALUATION OF PORV OPENING PROBABILITY DURING OVERPRESSURE TRANSIENT

DESCRIPTION

This NUREG-0660[48] item required all B&W operating plants (OLs and OL applicants) to document that their PORVs would open in less than 5% of all anticipated overpressure transients. Clarifications were issued in NUREG-0737[98] to include all PWRs.

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.3(8): FURTHER STAFF CONSIDERATION OF NEED FOR DIVERSE DECAY HEAT REMOVAL METHOD INDEPENDENT OF SGs

DESCRIPTION

This NUREG-0660[48] item affected all PWRs (OLs and OL applicants). However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Items II.C.1 and II.E.3.3.

CONCLUSION

This item is covered in Items II.C.1 and II.E.3.3.

ITEM II.K.3(9): PROPORTIONAL INTEGRAL DERIVATIVE CONTROLLER MODIFICATION

DESCRIPTION

This NUREG-0660[48] item required all W plants (OLs and OL applicants) to raise the interlock bistable trip setting to preclude derivative action from opening the PORVs. Clarifications affecting both groups of plants were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-40 was established by DL for implementation purposes.

ITEM II.K.3(10): ANTICIPATORY TRIP MODIFICATION PROPOSED BY SOME LICENSEES TO CONFINE RANGE OF USE TO HIGH POWER LEVELS

DESCRIPTION

This NUREG-0660[48] item required that the anticipatory trip modification proposed by some licensees to confine the range of use of high-power levels not be made until it could be shown that the probability of a small-break LOCA resulting from a stuck-open PORV was substantially unaffected by the modification. The applicability of the item to W operating plants and OL applicants with W reactors was to be determined on a plant-by-plant basis. Clarifications affecting both groups of plants were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-41 was established by DL for implementation purposes.

ITEM II.K.3(11): CONTROL USE OF PORV SUPPLIED BY CONTROL COMPONENTS, INC. UNTIL FURTHER REVIEW COMPLETE

DESCRIPTION

This NUREG-0660[48] item required plants to justify the use of PORVs that had failed during testing. The applicability of the item to all operating plants and OL applicants was to be determined on a case-by-case basis. Clarifications affecting both groups of plants were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.3(12): CONFIRM EXISTENCE OF ANTICIPATORY TRIP UPON TURBINE TRIP

DESCRIPTION

This NUREG-0660[48] item required all W plants (OLs and OL applicants) to confirm that their plants have an anticipatory reactor trip upon turbine trip. Plants that did not have this trip were required to provide a conceptual design and evaluation for the installation of the trip. Clarifications affecting both groups of plants were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-42 was established by DL for implementation purposes.

ITEM II.K.3(13): SEPARATION OF HPCI AND RCIC SYSTEM INITIATION LEVELS

DESCRIPTION

This NUREG-0660[48] item required all GE plants (OLs and OL applicants) to analyze the benefits to be gained from separating HPCI and RCIC initiation levels and providing auto-start of RCIC on low-low level. Clarifications were issued in NUREG-0737[98] to include all operating BWRs and OL applicants with RCIC and HPCI systems.

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-43 was established by DL for implementation purposes.

ITEM II.K.3(14): ISOLATION OF ISOLATION CONDENSERS ON HIGH RADIATION

DESCRIPTION

This NUREG-0660[48] item required all operating GE plants with isolation condensers to increase the availability of the isolation condensers as heat sinks by providing high radiation isolation signals at the vent rather than at the steam lines. Clarifications affecting all operating BWRs with isolation condensers were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-44 was established by DL for implementation purposes.

ITEM II.K.3(15): MODIFY BREAK DETECTION LOGIC TO PREVENT SPURIOUS ISOLATION OF HPCI AND RCIC SYSTEMS

DESCRIPTION

This NUREG-0660[48] item required all GE plants (OLs and OL applicants) to modify their pipe break detection circuitry to prevent isolation of system(s) due to startup pressure transient. Clarifications were issued in NUREG-0737[98] to address all BWRs (OLs and OL applicants) with HPCI and RCIC systems.

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-45 was established by DL for implementation purposes.

ITEM II.K.3(16): REDUCTION OF CHALLENGES AND FAILURES OF RELIEF VALVES - FEASIBILITY STUDY AND SYSTEM MODIFICATION

DESCRIPTION

This NUREG-0660[48] item required all GE plants (OLs and OL applicants) to study the reduction in challenge and failure rates of relief valves to minimize the most possible cause of a small-break LOCA. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-46 was established by DL for implementation purposes.

ITEM II.K.3(17): REPORT ON OUTAGE OF ECC SYSTEMS - LICENSEE REPORT AND TECHNICAL SPECIFICATION CHANGES

DESCRIPTION

This NUREG-0660[48] item required all GE plants (OLs and OL applicants) to review data on ECC system outages to determine if cumulative outage time limitations should be incorporated in technical specifications. Clarifications were issued in NUREG-0737[98] to include all operating reactors and OL applicants.

CONCLUSION

This item was RESOLVED, requirements were issued and MPA F-47 was established by DL for implementation purposes.

ITEM II.K.3(18): MODIFICATION OF ADS LOGIC - FEASIBILITY STUDY AND MODIFICATION FOR INCREASED DIVERSITY FOR SOME EVENT SEQUENCES

DESCRIPTION

This NUREG-0660[48] item required all GE plants (OLs and OL applicants) to modify their ADS actuation logic to eliminate the need for manual actuation to assure adequate core cooling. A feasibility study and risk assessment study were required to determine the optimum approach. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-48 was established by DL for implementation purposes.

ITEM II.K.3(19): INTERLOCK ON RECIRCULATION PUMP LOOPS

DESCRIPTION

This NUREG-0660[48] item required all GE operating plants with non-jet pumps to install interlocks to assure that level measurements are representative of the level in the core. Clarifications were issued in NUREG-0737[98] to address all operating BWRs with non-jet pumps, except Humboldt Bay.

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-49 was established by DL for implementation purposes.

ITEM II.K.3(20): LOSS OF SERVICE WATER FOR BIG ROCK POINT

DESCRIPTION

This NUREG-0660[48] item required Big Rock Point to evaluate the acceptability or the consequences of a loss of service water. A clarification to this requirement was issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED and requirements were issued.

ITEM II.K.3(21): RESTART OF CORE SPRAY AND LPCI SYSTEMS ON LOW LEVEL - DESIGN AND MODIFICATION

DESCRIPTION

This NUREG-0660[48] item required all GE plants (OLs and OL applicants) to modify their core spray and LPCI system logic so that these systems would restart, if required, to assure adequate core cooling. It was believed that the core spray and LPCI system flow may be stopped by the operator. These systems could not start automatically on loss of water level if an initiation signal were still present. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-50 was established by DL for implementation purposes.

ITEM II.K.3(22): AUTOMATIC SWITCHOVER OF RCIC SYSTEM SUCTION - VERIFY PROCEDURES AND MODIFY DESIGN

DESCRIPTION

This NUREG-0660[48] item affected all GE plants (OLs and OL applicants). The RCIC system takes suction from the condensate storage tank with manual switchover to the suppression pool when the condensate storage tank level is low. This switchover should be made automatically. Until the automatic switchover is implemented, licensees should verify that clear and cogent procedures exist for the manual switchover of the RCIC system suction from the condensate storage tank to the suppression pool. Clarifications affecting all operating BWRs and OL applicants with RCIC systems were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-51 was established by DL for implementation purposes.

ITEM II.K.3(23): CENTRAL WATER LEVEL RECORDING

DESCRIPTION

This NUREG-0660[48] item was originated to address GE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660[48], it was determined that the issue was being addressed by Items I.D.2, III.A.1.2, and III.A.3.4.

CONCLUSION

This item is covered in Items I.D.2, III.A.1.2, and III.A.3.4.

ITEM II.K.3(24): CONFIRM ADEQUACY OF SPACE COOLING FOR HPCI AND RCIC SYSTEMS

DESCRIPTION

This NUREG-0660[48] item required all operating GE plants (OLs and OL applicants) to verify that HPCI and RCIC are designed to withstand loss of offsite power for at least 2 hours. Clarifications affecting all BWRs (OLs and OL applicants) with HPCI and RCIC systems were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-52 was established by DL for implementation purposes.

ITEM II.K.3(25): EFFECT OF LOSS OF AC POWER ON PUMP SEALS

DESCRIPTION

This NUREG-0660[48] item required all GE plants (OLs and OL applicants) to verify the adequacy of pump seals to withstand loss of cooling water due to loss of AC power for at least 2 hours. Clarifications were issued in NUREG-0737[98] to include all BWRs, W and CE operating reactors, and all OL applicants.

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-53 was established by DL for implementation purposes.

ITEM II.K.3(26): STUDY EFFECT ON RHR RELIABILITY OF ITS USE FOR FUEL POOL COOLING

DESCRIPTION

This NUREG-0660[48] item was originated to affect GE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Item II.E.2.1.

CONCLUSION

This item is covered in Item II.E.2.1.

ITEM II.K.3(27): PROVIDE COMMON REFERENCE LEVEL FOR VESSEL LEVEL INSTRUMENTATION

DESCRIPTION

This NUREG-0660[48] item affected all GE plants (OLs and OL applicants) and required all reactor vessel level instruments to be referenced to the same point. It was believed that different reference points of the various reactor vessel water level instruments could cause operator confusion. Either the bottom of the vessel or the active fuel were considered to be reasonable reference points. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-54 was established by DL for implementation purposes.

ITEM II.K.3(28): STUDY AND VERIFY QUALIFICATION OF ACCUMULATORS ON ADS VALVES

DESCRIPTION

This NUREG-0660[48] item affected all plants with GE reactors (OLs and OL applicants). These plants were required to assure that air or nitrogen accumulators for ADS valves had sufficient capacity to cycle the valves open five times at design pressure. However, clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-55 was established by DL for implementation purposes.

ITEM II.K.3(29): STUDY TO DEMONSTRATE PERFORMANCE OF ISOLATION CONDENSERS WITH NON-CONDENSIBLES

DESCRIPTION

This NUREG-0660[48] item affected all operating plants with GE isolation condensers. These plants were required to demonstrate the adequacy of isolation condensers with non-condensibles. Clarifications affecting all operating BWRs with isolation condensers were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-56 was established by DL for implementation purposes.

ITEM II.K.3(30): REVISED SMALL-BREAK LOCA METHODS TO SHOW COMPLIANCE WITH 10 CFR 50, APPENDIX K

DESCRIPTION

This NUREG-0660[48] item required all OLs and OL applicants to revise and submit for NRC approval the analyses used by NSSS vendors and/or fuel suppliers for SBLOCA analysis in compliance with 10 CFR 50, Appendix K. The revised analyses were to account for comparisons with experimental data, including data from the LOFT and semiscale test facilities. Clarifications were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-57 was established by DL for implementation purposes.

ITEM II.K.3(31): PLANT-SPECIFIC CALCULATIONS TO SHOW COMPLIANCE WITH 10 CFR 50.46

DESCRIPTION

This NUREG-0660[48] item required all OLs and OL applicants to submit for NRC approval plant-specific calculations using NRC-approved models for SBLOCA, to show compliance with 10 CFR 50.46. Clarifications were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-58 was established by DL for implementation purposes.

ITEM II.K.3(32): PROVIDE EXPERIMENTAL VERIFICATION OF TWO-PHASE NATURAL CIRCULATION MODELS

DESCRIPTION

This NUREG-0660[48] item was originated to require PWRs to provide experimental verification of two-phase natural circulation models. However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Item II.E.2.2.

CONCLUSION

This item is covered in Item II.E.2.2.

ITEM II.K.3(33): EVALUATE ELIMINATION OF PORV FUNCTION

DESCRIPTION

This NUREG-0660[48] item was originated to require PWRs (OLs and OL applicants) to evaluate elimination of the PORV function. However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Item II.C.1.

CONCLUSION

This item is covered in Item II.C.1.

ITEM II.K.3(34): RELAP-4 MODEL DEVELOPMENT

DESCRIPTION

This NUREG-0660[48] item was originated to address RELAP-4 model development in PWRs. However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Item II.E.2.2.

CONCLUSION

This item is covered in Item II.E.2.2.

ITEM II.K.3(35): EVALUATION OF EFFECTS OF CORE FLOOD TANK INJECTION ON SMALL-BREAK LOCAs

DESCRIPTION

This NUREG-0660[48] item was originated to evaluate the effects of core flood tank injection on SBLOCAs in B&W plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Item I.C.1.

CONCLUSION

This item is covered in Item I.C.1.

ITEM II.K.3(36): ADDITIONAL STAFF AUDIT CALCULATIONS OF B&W SMALL-BREAK LOCA ANALYSES

DESCRIPTION

This NUREG-0660[48] item was originated to address B&W plants, but was determined to be covered by Item I.C.1 prior to the publication of NUREG-0660.[48]

CONCLUSION

This item is covered in Item I.C.1.

ITEM II.K.3(37): ANALYSIS OF B&W RESPONSE TO ISOLATED SMALL-BREAK LOCA

DESCRIPTION

This NUREG-0660[48] item was originated to analyze the response of B&W plants (OLs and OL applicants) to isolated SBLOCAs. However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Item I.C.1.

CONCLUSION

This item is covered in Item I.C.1.

ITEM II.K.3(38): ANALYSIS OF PLANT RESPONSE TO A SMALL-BREAK LOCA IN THE PRESSURIZER SPRAY LINE

DESCRIPTION

This NUREG-0660[48] item was originated to analyze the reponse of B&W plants (OLs and OL applicants) to a SBLOCA in the pressurizer spray line. However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Item I.C.1.

CONCLUSION

This item is covered in Item I.C.1.

ITEM II.K.3(39): EVALUATION OF EFFECTS OF WATER SLUGS IN PIPING CAUSED BY HPI AND CFT FLOWS

DESCRIPTION

This NUREG-0660[48] item was originated to evaluate the effects of water slugs caused by HPI and CFT flows in the piping of B&W plants (OLs and OL applicants).

However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Item I.C.1.

CONCLUSION

This item is covered in Item I.C.1.

ITEM II.K.3(40): EVALUATION OF RCP SEAL DAMAGE AND LEAKAGE DURING A SMALL-BREAK LOCA

DESCRIPTION

This NUREG-0660[48] item was originated to evaluate RCP seal damage and leakage during a SBLOCA in B&W plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Item II.K.2(16).

CONCLUSION

This item is covered in Item II.K.2(16).

ITEM II.K.3(41): SUBMIT PREDICTIONS FOR LOFT TEST L3-6 WITH RCPs RUNNING

DESCRIPTION

This NUREG-0660[48] item was originated to require B&W plants (OLs and OL applicants) to submit to the NRC predictions for LOFT Test L3-6 with RCPs running. However, prior to the publication of NUREG-0660,[48] it was determined that the issued was being addressed by Item I.C.1.

CONCLUSION

This item is covered in Item I.C.1.

ITEM II.K.3(42): SUBMIT REQUESTED INFORMATION ON THE EFFECTS OF NON-CONDENSIBLE GASES

DESCRIPTION

This NUREG-0660[48] item was originated to require B&W plants (OLs and OL applicants) to submit to the NRC requested information on the affects of non-condensible gases. However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Item I.C.1.

CONCLUSION

The item is covered in Item I.C.1.

ITEM II.K.3(43): EVALUATION OF MECHANICAL EFFECTS OF SLUG FLOW ON STEAM GENERATOR TUBES

DESCRIPTION

This NUREG-0660[48] item was originated to evaluate the mechanical affects of slug flow on the steam generator tubes of B&W plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Item II.K.2(15).

CONCLUSION

This item is covered in Item II.K.2(15).

ITEM II.K.3(44): EVALUATION OF ANTICIPATED TRANSIENTS WITH SINGLE FAILURE TO VERIFY NO SIGNIFICANT FUEL FAILURE

DESCRIPTION

This NUREG-0660[48] item required all GE plants (OLs and OL applicants) to show that transients combined with the worst single failure would not result in significant fuel damage. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-59 was established by DL for implementation purposes.

ITEM II.K.3(45): EVALUATE DEPRESSURIZATION WITH OTHER THAN FULL ADS

DESCRIPTION

This NUREG-0660[48] item required all GE plants (OLs and OL applicants) to analyze depressurization modes other than full ADS for possible inclusion in emergency procedures. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-60 was established by DL for implementation purposes.

ITEM II.K.3(46): RESPONSE TO LIST OF CONCERNS FROM ACRS CONSULTANT

DESCRIPTION

This NUREG-0660[48] item required all GE plants (OLs and OL applicants) to respond to concerns raised by ACRS consultants. Clarifications affecting all BWRs (OLs and OL applicants) were issued in NUREG-0737.[98]

CONCLUSION

This item is RESOLVED, requirements were issued, and MPA F-61 was established by DL for implementation purposes.

ITEM II.K.3(47): TEST PROGRAM FOR SMALL-BREAK LOCA MODEL VERIFICATION PRETEST PREDICTION, TEST PROGRAM, AND MODEL VERIFICATION

DESCRIPTION

This NUREG-0660[48] item was originated to require GE plants (OLs and OL applicants) to complete a test program for SBLOCA model verification. However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Items I.C.1 and II.E.2.2.

CONCLUSION

This item is covered in Items I.C.1 and II.E.2.2.

ITEM II.K.3(48): ASSESS CHANGE IN SAFETY RELIABILITY AS A RESULT OF IMPLEMENTING B&OTF RECOMMENDATIONS

DESCRIPTION

This NUREG-0660[48] item was originated to require GE plants (OLs and OL applicants) to assess the change in safety reliability as a result of implementing the recommendations of the Bulletins and Orders Task Force (B&OTF). However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Items II.C.1 and II.C.2.

CONCLUSION

This item is covered in Items II.C.1 and II.C.2.

ITEM II.K.3(49): REVIEW OF PROCEDURES (NRC)

DESCRIPTION

This NUREG-0660[48] item was originated to address all W and CE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Item I.C.9 for OLs and by Items I.C.8 and I.C.9 for OL applicants.

CONCLUSION

This item is covered in Items I.C.8 and I.C.9.

ITEM II.K.3(50): REVIEW OF PROCEDURES (NSSS VENDORS)

DESCRIPTION

This NUREG-0660[48] item was originated to address all W and CE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Item I.C.9 for OLs and by Items I.C.7 and I.C.9 for OL applicants.

CONCLUSION

This item is covered in Items I.C.7 and I.C.9.

ITEM II.K.3(51): SYMPTOM-BASED EMERGENCY PROCEDURES

DESCRIPTION

This NUREG-0660[48] item was originated to address all W, CE, and GE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Item I.C.9.

CONCLUSION

This item is covered in Item I.C.9.

ITEM II.K.3(52): OPERATOR AWARENESS OF REVISED EMERGENCY PROCEDURES

DESCRIPTION

This NUREG-0660[48] item was originated to address all GE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Items I.B.1.1, I.C.2, and I.C.5.

CONCLUSION

This item is covered in Items I.B.1.1, I.C.2, and I.C.5.

ITEM II.K.3(53): TWO OPERATORS IN CONTROL ROOM

DESCRIPTION

This NUREG-0660[48] item was originated to address all GE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Item I.A.1.3.

CONCLUSION

This item is covered in Item I.A.1.3.

ITEM II.K.3(54): SIMULATOR UPGRADE FOR SMALL-BREAK LOCAs

DESCRIPTION

This NUREG-0660[48] item was originated to address all plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Item I.A.4.1.

CONCLUSION

This item is covered in Item I.A.4.1.

ITEM II.K.3(55): OPERATOR MONITORING OF CONTROL BOARD

DESCRIPTION

This NUREG-0660[48] item was originated to address all W and CE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Items I.C.1, I.D.2, and I.D.3.

CONCLUSION

This item is covered in Items I.C.1, I.D.2, and I.D.3.

ITEM II.K.3(56): SIMULATOR TRAINING REQUIREMENTS

DESCRIPTION

This NUREG-0660[48] item was originated to address all W and CE plants (OLs and OL applicants). However, prior to the publication of NUREG-0660,[48] it was determined that the issue was being addressed by Items I.A.2.6 and I.A.3.1.

CONCLUSION

This item is covered in Items I.A.2.6 and I.A.3.1.

ITEM II.K.3(57): IDENTIFY WATER SOURCES PRIOR TO MANUAL ACTIVATION OF ADS

DESCRIPTION

This NUREG-0660[48] item required all operating GE plants to revise their emergency procedures to include verification that low pressure cooling systems are available prior to manual ADS. For OL applicants, the issue was determined to be covered by Item I.C.1. Clarifications affecting all operating BWRs were issued in NUREG-0737.[98]

CONCLUSION

This item was RESOLVED, requirements were issued, and MPA F-62 was established by DL for implementation purposes.

REFERENCES

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980.

93. NUREG-0611, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants," U.S. Nuclear Regulatory Commission, January 1980.

94. NUREG-0626, "Staff Report on the Generic Assessment of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Boiling Water Reactors Designed by the General Electric Company," U.S. Nuclear Regulatory Commission, January 1980.

95. NUREG-0635, "Generic Assessment of Small Break Loss-of-Coolant Accidents in Combustion Engineering Designed Operating Plants," U.S. Nuclear Regulatory Commission, January 1980.

96. NUREG-0565, "Staff Report on the Generic Evaluation of Small-Break Loss-of-Coolant Accident Behavior for Babcock and Wilcox Operating Plants," U.S. Nuclear Regulatory Commission, January 1980.

97. NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small-Break Loss-of-Coolant Accidents in Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, December 1979.

98. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980.

578. NUREG-0748, "Operating Reactors Licensing Actions Summary," U.S. Nuclear Regulatory Commission.

579. NUREG-0694, "TMI-Related Requirements for New Operating Licenses," U.S. Nuclear Regulatory Commission, June 1980.

580. NUREG-0645, "Report of the Bulletins and Orders Task Force," U.S. Nuclear Regulatory Commission, January 1980.

ITEM A-41: LONG-TERM SEISMIC PROGRAM

DESCRIPTION

Historical Background

In a memorandum[127] dated June 7, 1976, NRR recommended that a study be initiated on the quantification of inherent seismic safety margins in NRR's seismic design requirements. This memo suggested the initiation of a long-term research program and outlined a long list of informational deficiencies to be addressed by this program. This program was included in NUREG-0371.[2]

Subsequently, a User Need Request[128] was forwarded to RES detailing the NRR needs.[127] In response to this request,[129] RES established a contract with LLNL to conduct the Seismic Safety Margins Research Program (SSMRP) to start in 1978. The program plan[130] was intended to provide the methodology to determine safety margins in a nuclear power plant subjected to a large earthquake. The objectives of the SSMRP are to estimate the conservatisms (or lack of conservatisms) in the SRP[11] seismic safety requirements and to develop improved requirements. The approach to achieve these objectives is to develop probabilistic methodology that enables one to more realistically predict the behavior of nuclear power plants safety systems, components, and structures during an earthquake. The SSMRP project in its first phase was divided into two major task areas: (1) calculation of major structural and subsystem responses, and (2) development of a code to calculate accident sequence probabilities (SEISM) tied to specific radioactive release categories. The final report for the first phase is contained in NUREG/CR-2015,[131] Vol. 1. Subsequent to the completion of the first phase of the SSMRP, an NRR memo[132] was issued to RES in which a significant redirection of the SSMRP was suggested in order to more directly fit NRR needs. Recommendations[132] were developed in conjunction with a recent review of the SSMRP and the Long Range Research Plan[133] of RES, together with discussions between the RES staff and the NRR staff. In general, the recommendations were based on the view that, while NRR's user needs and programmatic goals for additional and follow-on seismic analysis research were the same as for SSMRP, it was becoming clear that significant advancements in seismic analysis and confidence in estimates of seismic structural risk cannot be achieved without improved and definitive data in the following areas: seismic input, soil structure interactions, dynamic structural response, and structural fragility. These revised needs were cognizant of the potentially significant changes inherent in SECY-82-53[134] in which the magnitude of the controlling earthquake in the eastern U.S. could increase. This was based on the possible modification of the U.S. Geological Survey position on the association of the 1886 Charlestown, SC, earthquake with geologic structure and the recent earthquakes in New Brunswick, Canada. The SSMRP was scheduled to end at the close of FY 1984.

Safety Significance

Recent PRA studies[187] have indicated that the seismic risk may be a significant contributor to the total risk for nuclear power plants. Most PRAs prepared to date do not include an assessment of risk from earthquakes. Thus, it is important that the NRC have methods to quantify and assess seismic risk to evaluate and enhance the credibility of PRAs. Also, there is a need to reexamine the traditional process of seismic analysis and design of nuclear power plants in an overall system context. This need comes principally from the widely held belief that a compounding of conservatisms occurs in the current process, i.e., at each stage of the current process, conservatisms are introduced to account for uncertainties and these conservatisms compound from one stage to the next. However, in each stage only minimal compensations are made for the compounding of conservatisms because they are not quantified. For example, the earthquake used in the seismic design represents the maximum earthquake potential (SSE) considering the geology, seismology, and specific characteristics of the subsurface material. The earthquake motion is coupled to the bedrock and building foundation through the use of conservative soil properties to produce the highest responses (forces and stresses). Such responses are compared to conservative estimates of the strength or capacity of each structure or component.

On the other hand, there is concern that the current licensing criteria may produce seismic designs that are apparently conservative for some features, but can have adverse effects on the overall plant safety. For example, piping made stiff to resist seismic loads may cause higher thermal expansion stresses in nozzles during normal operation.

Thus, NRC has established regulations, guides, and licensing review procedures that define seismic safety criteria for nuclear power plant design. These criteria collectively constitute a seismic methodology chain (SMC). The seismic safety criteria for nuclear power plant design were developed to ensure structural integrity and functional safety of buildings, equipment, and components. They depart from the conventional earthquake engineering practice in detail and complexity. The overall SMC is considered sufficiently conservative to ensure safety. However, it is thought to be necessary to characterize the overall seismic safety and to improve it by establishing new criteria as may be required.

CONCLUSION

The NRC must be prepared to provide the basis for licensing decisions involving operating plants that are required to consider changing seismic loads and design criteria. By knowing and understanding the inherent conservatisms in the seismic design (i.e., being able to more accurately characterize the realistic behavior of structures and components under earthquake conditions), the NRC would be better able to judge the necessity and extent of modifying and requalifying structures and components in older operating plants to be reviewed for increased seismic loads or of improving design criteria for new standardized plants.

The SSMRP is also tied to systems research involving PRA such as the IREP/NREP effort. The SSMRP will provide the seismic risk methodology that is currently lacking in these programs.

Reliable estimates on how much the seismic risk would change as a result of the completion of this program were not obtained because the frequencies and magnitudes of earthquakes are uncertain and the failure probabilities (i.e., fragility) cannot be inferred directly from the objectives and expected results of this program. However, because this program has a direct bearing on Item B-6 (Loads, Load Combinations, Stress Limits), which has a high priority ranking, as well as this program's relationship to PRA, plant costs, and overall plant safety, the issue was given a medium priority ranking.

However, a reevaluation of this issue by DE in October 1984 revealed that the programs covered by the issue were intended to gather and develop information; there were no plans to revise regulatory requirements upon completion of the programs. It was determined that the programs were long range on-going activities jointly sponsored by NRR and RES and were being adequately tracked.[692] Thus, this issue was RESOLVED and no new requirements were established.

REFERENCES

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.

11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.

127. Memorandum for H. Kouts from B. Rusche, "Quantification of Inherent Safety Margins in Seismic Design (SAFER-76-5)," June 7, 1976.

128. Memorandum for S. Levine from E. Case, "Quantification of Inherent Safety Margins in Seismic Design," June 17, 1977.

129. Memorandum for H. Denton from S. Levine, "RES Response to NRR User Request on Quantification of Inherent Safety Margins to Seismic Design," November 1, 1978.

130. Memorandum for S. Levine from H. Denton, "Seismic Safety Margins Research Program," February 23, 1979.

131. NUREG/CR-2015, "SSMRP Phase 1 Final Report," U.S. Nuclear Regulatory Commission, June 1982.

132. Memorandum for R. Minogue from H. Denton, "NRR Research Needs in Seismic Analysis Methodology," April 8, 1982.

133. NUREG-0784, "Long Range Research Plan, FY 1984-1988," U.S. Nuclear Regulatory Commission, August 1982.

134. SECY-82-52, "Possible Relocation of Design Controlling Earthquakes in the Eastern U.S.," U.S. Nuclear Regulatory Commission, February 5, 1980.

187. NUREG/CR-2300, "PRA Procedures Guide," U.S. Nuclear Regulatory Commission, September 1981.

692. Memorandum for T. Speis from H. Denton, "Generic Issue A-41; 'Long Term Seismic Program,'" October 10, 1984.

ITEM B-10: BEHAVIOR OF BWR MARK III CONTAINMENTS

DESCRIPTION

Historical Background

The description of this item given in NUREG-0471[3] is as follows:

> "This is an ACRS generic concern. Evaluation and approval is required of various aspects of the MARK III containment design which differs from the previously reviewed MARK I and MARK II designs. The task involves the completion of the staff evaluation of the MARK III containment and documentation of the method used to validate the analytical models and assumptions needed to predict the containment pressures in the event of a LOCA."

The MARK III suppression pool dynamic loads were reviewed by the NRC at the CP stage for Grand Gulf Nuclear Station, Units 1 and 2, and at the preliminary design analysis (PDA) stage for GESSAR-238NI. It was concluded at the time that the information available was sufficient to adequately define the pool dynamic loads for those nuclear plants under review for CPs. Since the issuance of the GESSAR-238NI SER in December 1975, GE has conducted further tests and analyses to confirm and refine the original load definitions. To keep the NRC and MARK III applicants apprised of the current status of these tests, GE issued an Interim Containment Loads Report (22A4365) in April 1978 and revised this report several times before GESSAR-II was provided to the NRC staff in March 1980. GESSAR-II is GE's FDA submittal for their standard BOP design and is to be referenced by MARK III OL applicants. Appendix 3B of GESSAR-II provides the finalized pool dynamic load definition for MARK III containments and is the basic document used for review by the NRC staff and its consultants.

The NRC staff is currently reviewing GE's pool dynamic load definitions to arrive at a finalized hydrodynamic load definition that can be utilized by MARK III containment applicants for operating licenses. The pool dynamic loads were being reviewed under USI A-39, "Determination of Safety Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containment." The end product of these two generic programs will be applicable to Grand Gulf.

Safety Significance

Following a postulated LOCA, escaping steam forces the suppression pool out of the drywell into the wet well. This action results in pool swell and loads from vent clearing, jets, chugging, impact of water, impact from froth impingement, pool fallback, condensation, and containment pressure.

The concern is that these loadings may damage structures and components located within the wet well. Although many of these structures (e.g., walkways) are by themselves not related to safety, the various ECCS systems take suction from the wet well and, therefore, damage in the wet well may affect the performance of the ECCS.

Possible Solution

The MARK III plants affected by this issue will be reviewed to determine if their structures meet the NRC Acceptance Criteria for MARK III LOCA-related pool dynamic loads. Structural fixes will then be implemented where necessary.

PRIORITY DETERMINATION

The assessment of this issue and its proposed resolution is based in part on an investigation performed by PNL.[64]

Frequency Estimate

The proposed resolution of the issue is the implementation of the structural fixes mentioned above. The applicable plants include all GE BWR/6 MARK III containments beginning with Grand Gulf 1 and 2. There are a total of 17 such plants listed in GESSAR II, Table 1.4-1, of which 4 have been cancelled, leaving a total of 13 applicable plants for forward-fit (i.e., fixes to be made before plant start-up). The Grand Gulf plant is selected as the representative plant. It is a typical GE BWR/6 with a MARK III Containment.

The parameters in the plant risk equations assumed to be affected by the BWR MARK III containment modifications are those related to the emergency core cooling system. The LOCA is taken to have already occurred, i.e., the dynamic loads are a result of the LOCA. The parameters are then selected from those which tend to mitigate the LOCA effects. The assumptions for the base case are:

(1) A LOCA occurs

(2) Some piping, equipment, or walkways are dislodged and fall onto the suction piping for the ECCS located in the suppression pool, plugging it in some manner. The plugging scenario described above is judged to have a probability of 10% for each suction. It affects elements L, LA2, LB2, LC, SA, and SB each of which becomes augmented by the amount 0.1. (The L sequences correspond to loss of suction for the various ECCS trains; SA and SB correspond to the suppression pool makeup system.)

The assumption for the adjusted case is that, after the postulated LOCA, no damage occurs to the ECCS piping because of structural fixes and, therefore, the originally calculated dominant accident sequences and frequencies prevail.[87]

Consequence Estimate

The affected sequences lead to a spectrum of consequences. The summed probability for Release Category BWR-2 is affected most, but the probabilities associated with the other three core-melt release categories are also affected. When the release category consequences are multiplied by their changes in frequency and then summed, the resultant risk reduction is 1,930 man-rem/RY. Based on an average remaining life of 30 years for the 13 affected plants, the total risk reduction associated with this issue is 7.5×10^5 man-rem.

Cost Estimate

Industry Cost: The structural fixes required to resist the LOCA-related pool dynamic loads at Grand Gulf 1 and 2 were selected as typical fixes for the generic issue and evaluated for cost. These included:

(1) Deleting solid circumferential concrete floor and adding a steel grating catwalk at the same elevation due to pool swell, plus relocating equipment to a higher elevation
(2) Relocating and strengthening main steam tunnel floor above pool swell zone
(3) Adding suppression pool makeup system
(4) Projecting TIP station floor down into suppression pool to eliminate pool swell loads
(5) Relocating piping to the region above bulk pool swell
(6) Changing piping submerged in pool to smaller sizes and heavier wall to accommodate submerged structure loads.

It is estimated that the modifications will cost \$2.6M plus \$10,000/year in extra maintenance for 30 years, giving a total cost of \$2.9M/plant for 13 plants. Therefore, the total industry cost is \$37.7M.

NRC Cost: NRC costs for review and inspection are estimated to be \$1.4M for the 13 plants.

Value/Impact Assessment

Based on a total risk reduction of 7.5×10^5 man-rem and a cost of \$39.1M, the value/impact score given by:

$$S = \frac{7.5 \times 10^5 \text{ man-rem}}{\$39.1\text{M}}$$

$$= 19{,}000 \text{ man-rem/\$M}.$$

CONCLUSION

This issue addresses the design adequacy and, therefore, the availability of containment, one level of the "defense-in-depth." Based on the value/impact score, the issue was identified as high priority. However, since the prioritization was completed, a value/impact analysis was published in NUREG-0978[600] and resulted in Revision 6 to SRP[11] Section 6.2.1.1.C. Thus, this item has been RESOLVED and new requirements were issued.[601]

REFERENCES

3. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," U.S. Nuclear Regulatory Commission, June 1978.

11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

87. Memorandum for T. Speis from R. Tedesco, Attachment, "Containment Systems Branch Input to the Safety Evaluation Report, Grand Gulf Nuclear Station, Units 1 and 2, Docket Numbers 50-416/417," March 25, 1982.

600. NUREG-0978, "Mark III LOCA-Related Hydrodynamic Load Definition," U.S. Nuclear Regulatory Commission, February 1984.

601. Memorandum for T. Combs from H. Denton, "Revised SRP Section 6.2.1.1.C of NUREG-0800," September 10, 1984.

ITEM B-26: STRUCTURAL INTEGRITY OF CONTAINMENT PENETRATIONS

DESCRIPTION

Historical Background

As described in NUREG-0471,[3] this issue involves staff evaluations to assess the adequacy of specific containment penetration designs from the point of view of structural integrity, ISI requirements, and new surveillance or analysis methods applicable to containment penetrations which are identified as inaccessible. The issue is applied to all operating plants as well as those plants currently under construction and up for licensing review.

In accordance with a DE memorandum,[215] that part of the issue involved with the structural integrity of specific containment penetration design, i.e., forged versus welded design, has been resolved. This resolution is based on a draft report by an NRC consultant. A NUREG is being considered to document this resolution. The second concern which involves the volumetric examination as required by ASME Code, Section XI[14] is only partially resolved for: (1) plants under licensing review, where inspection and surveillance problems associated with inaccessible penetrations must be resolved in some manner before startup operations can occur, and (2) operating reactors, where inspection and surveillance problems are reviewed during reviews of licensees' ISI programs.

The staff review should determine whether or not the configuration and accessibility of the welds in the proposed design and the procedures proposed for performing volumetric examination permit inservice examination requirements of Section XI[14] of the ASME Code at an augmented frequency in break exclusion regions, as required by SRP[11] Section 3.6.2. In the event that penetration designs are found inadequate with respect to conducting current inservice inspections, alternative surveillance or analysis methods would be implemented to ensure that inspections can be completed.[215]

Safety Significance

Upon satisfactory resolution of inspectability concerns, this issue should not affect public risk. However, should it be impractical for a plant to assure the above stated inservice examination requirement in accordance with SRP[11] Section 3.6.2, no specific guidance is provided as to what measures provide an acceptable resolution. In these cases, staff approval on a case-by-case review basis may result in inconsistent penetration requirements from plant to plant. Such inconsistencies, should they occur, could result in increased risk to the public. To account for this possibility, the potential public risk reduction is obtained by assuming that the likelihood of radioactive releases from containment may be reduced.

Possible Solution

The specific containment penetrations involved in this issue include only the high-energy fluid systems. High-energy fluid systems are defined as those that are in operation or pressurized during normal plant conditions (i.e., during reactor startup, power operation, and reactor cold shutdown excluding test modes) where either or both of the following criteria are satisfied: (1) maximum temperature exceeds 200°F, and (2) maximum pressure exceeds 275 psig.

For those plants or penetrations that do not or cannot meet the above inservice examination requirements, the staff should develop guidelines and/or criteria to provide consistent requirements and acceptable conditions to assure the acceptability of the penetration designs and minimum levels of inspectability to meet these criteria.

PRIORITY DETERMINATION

Assumptions

The PNL analysis[64] for this issue assumed that all penetration assembly designs meet code accessibility requirements or approved analysis/surveillance techniques. The result is adequate completion of ISI as well as elimination of unresolved conditions affecting plant startup.

An average of 40 high-energy penetrations per plant are assumed in the following analysis. This number will vary depending on plant type and design and is only an estimate based on information available in several BWR and PWR FSARs (including tables of high-energy lines, identification of systems requiring boundary guard pipes and complete listings of penetration data). It is further assumed that only 20% (8) of all high-energy penetrations per plant need attention as specified by the issue. Since requirements for ISI are known, industry, where possible, attempts to build in inspectability features.

There are analysis and augmented inspection procedures currently available to accommodate many of the inaccessible penetrations. It is estimated that 20% of the 8 penetrations under consideration may require the development of new analysis procedures. Therefore, the number of penetrations per plant requiring new procedures is (0.2)(8) or approximately 2 penetrations per plant. Of the originally assumed 40 penetrations, the 2 penetrations per plant requiring new procedures would be 5 times more likely to fail than the remaining 38. Upon resolution of the issue, all 40 penetrations would have an equal failure probability. This results in a 17% reduction in the containment leakage probability.

Frequency Estimate

For those plants that meet the current requirements (SRP[11] Section 3.6.2), this issue results in no change in the core-melt frequency. To determine the potential effect on core-melt frequency associated with inadequate containment penetration designs, the containment failure mode B (containment leakage) is assumed.

For PWRs, PNL[64] selected the Oconee 3 reactor as their representative model. The base case core-melt frequencies for the PWR-4 and PWR-5 releases were 9.7×10^{-8}/RY and 4.6×10^{-7}/RY, respectively. The reduced core-melt frequencies for PWR-4 and PWR-5 type releases were 7.9×10^{-8}/RY and 3.8×10^{-7}/RY, respectively.

For BWRs, PNL used the Grand Gulf reactor with a BWR-4 release as the representative model. The base case core-melt frequency was determined to be 2.4×10^{-7}/RY. The potential reduced BWR core-melt frequency was 2×10^{-7}/RY.

Consequence Estimate

Assuming PWR-4 and PWR-5 releases result in 2.7×10^{6} man-rem and 1×10^{6} man-rem, respectively, the potential risk reduction is 1.3×10^{-1} man-rem/RY for PWRs. The average remaining life for the 47 operating and 43 planned PWRs is 28.8 years. As a result, the potential public risk reduction considering 2,592 RY is 337 man-rem for all PWRs.

For BWRs with a potential BWR-4 release that results in 6.1×10^{5} man-rem, the risk reduction is 3×10^{-2} man-rem/RY. The average remaining life for 24 operating and 20 planned BWRs is 27.4 years. The potential risk reduction considering 1,205 RY is 36 man-rem for all BWRs. Therefore, the total public risk reduction is 373 man-rem.

Cost Estimate

NRC Cost: Development of guidelines/criteria are assumed to take one man-year ($100,000). If this cost is divided among all operating and planned reactors (134 plants), the per plant cost is $750. Approximately 5 man-wks ($9,620) are currently required to complete a plant-specific review. The developed guidelines/criteria can be expected to reduce the plant-specific reviews to 3 man-wks/plant ($5,770/plant). Therefore, a net NRC cost benefit of approximately $3,100/plant is obtainable by development of guidelines/criteria for this issue.

Industry Cost: It is estimated that 8 man-wks/plant is currently needed to develop supporting analyses and procedures on a plant-specific basis. Appropriate guidelines/criteria can be expected to reduce this effort by 3 man-wks/plant ($5,700/plant). Assuming that the guidelines/criteria may require new inspections or analysis every ten-year inspection period, an additional 4 man-wks/plant/10 years over an average remaining plant life of 30 years would result in an additional cost of $7,700/plant.

Considering the potential cost savings of $5,700 afforded by the guidelines and the potential cost (impact) of additional requirements ($7,700), the net cost (impact) is $2,000/plant. However, if the initial plant-specific reviews without the guidelines were to result in similar inspection requirements (which is likely), the above impact cost of $7,700 would be moot and the result would be a plant cost benefit of $5,700.

Based on the above cost estimates, the combined NRC and Industry Costs result in a net cost benefit (value) ranging from $1,000/plant to $9,000/plant.

Value/Impact Assessment

The values associated with the issue are: (1) a small potential public risk reduction of 373 man-rem, (2) a net NRC and industry cost benefit of $1,000 to $9,000 per affected plant, and (3) a potential reduction in ORE of 1,200 man-rem for the 63 plants not yet operational (see Other Considerations below). No impacts should result from development of guidelines/criteria for alternate surveillance or analysis methods for inaccessible penetrations.

Other Considerations

PNL estimated that at most one failure per year occurs in all (71) operating plants. The time to repair the failure involves about 20 man-wks/failure in a 0.25 R/hr field required (2.8 man-rem/RY).

No reduction in exposure will be credited to plants that are already designed and operating, since repairs would be predicted in existing designs, requirements, and failure rates. Development of the guidelines/criteria, as described in the above assumptions, were estimated to result in a 17% reduction in containment leakages. The potential reduction in ORE for the 63 planned reactors over a 40-year design life is [(0.17)(63)(40)(2.8)] man-rem or 1,200 man-rem.

CONCLUSION

Based on the above value/impact assessment, this issue was identified as medium priority. However, after a reevaluation of the issue by DE, it was concluded that further expenditure of resources was unwarranted. DE believed that the increase in ORE from additional inspections would negate the small potential risk reduction associated with the issue.[647] Thus, the issue was RESOLVED and no new requirements were established.

REFERENCES

3. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," U.S. Nuclear Regulatory Commission, June 1978.

11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.

14. ASME Boiler And Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers, 1974.

16. WASH-1400 (NUREG-75/014), "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, October 1975.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

215. Memorandum for E. Sullivan from R. Bosnak, "Generic Issues," September 17, 1982.

647. Memorandum for T. Speis from H. Denton, "Resolution of Generic Issue B-26, 'Structural Integrity of Containment Penetrations,'" September 27, 1984.

ITEM B-54: ICE CONDENSER CONTAINMENTS

DESCRIPTION

Historical Background

This NUREG-0471[3] item deals with two concerns regarding the ice condenser containment design. The first concern arises from an ACRS comment on the D. C. Cook Unit 1 review. The normal procedure used by the staff (CSB) to conclude on the adequacy of containment design entails performing a confirmatory analysis of the applicant's design basis accident and approving or disapproving the design on the basis of comparison of the two analyses. CSB uses the CONTEMPT-LT[17] Code developed by INEL to perform independent containment analyses. The CONTEMPT-LT[17] Code does not have the capability to analyze an ice condenser containment. The staff's review of the ice condenser design has, therefore, been conducted by rigorous review of the applicant's code, LOTIC (developed and used by Westinghouse as containment designer for applicants using the ice condenser containment design), and the full-scale ice condenser test program conducted by Westinghouse to prove their design. In their initial review of the D. C. Cook plant, the ACRS recommended that the staff develop a computer code with ice condenser capability in order to perform independent confirmatory calculations in the fashion normally utilized in containment design review.

The second part of this issue deals with technical specifications regarding the weighing of ice in the approximately 2,000 baskets in the ice condenser and the minimum acceptable ice weights and ice condenser inspection frequency. The concern has been spawned by the nonsymmetric ice losses by sublimation experienced at D. C. Cook and later at Sequoyah. Present technical specifications do not consider the patterns of ice loss experienced in the field and DOR was looking ahead anticipating requests for relief in the future.

Safety Significance

Both parts of the issue deal with the ice condenser capability to extract blowdown energy in the early phase (first hour) of a LOCA. After ice-melt, the containment pressure control is provided by containment spray systems for continued energy removal as in conventional dry containment designs. Failure of containment by overpressure due to inadequate ice inventory, inadequate ice condenser surveillance and maintenance, or faulty analysis would be the expected result. Two types of accidents could occur, each having a different expected frequency. The first would be a large LOCA with early containment failure but with adequate core cooling. The second event would be a large LOCA again with early containment failure followed by loss of core cooling and core-melt.

Possible Solution

Should a perception be reached that the probability of containment failure due to LOCA overpressurization was too high, peak containment pressure could be reduced by increasing the containment spray system capacity drastically. This would entail as much as a fivefold increase in the spray system flow capacity. It would probably require complete new redundant containment spray systems.

PRIORITY DETERMINATION

Frequency Estimate

The containment spray system in plants with ice condenser containment is conservatively sized to prevent containment pressure greater than design pressure after ice-melt for an assumed large-break LOCA. Ice-melt is calculated and predicted by the full-scale ice condenser test program to occur, at the earliest, about 1 hour after a large-break LOCA. Containment failure is not predicted until containment pressure exceeds at least twice the design pressure. Therefore, the containment spray systems could handle the energy releases to containment after about the first half-hour after a LOCA without containment failure. This means the ice condenser must be effective for the first half-hour after a LOCA. During this time period, effective core cooling must take place or the decay energy will not reach the containment. The first half-hour after a large-break LOCA coincides roughly with ECCS operation in the injection mode. From WASH-1400,[16] the frequency of a large LOCA is 10^{-3}/RY to 10^{-4}/RY.

The ice condenser containment could fail during the first half-hour of a large LOCA due to overpressure if there were an analytical error in design or an inadequacy of technical specifications governing ice condenser operations. The analytical model (LOTIC)[167] has been checked and double-checked by Westinghouse, licensees, and the NRC. The same is true of the technical specifications. The probability of having errors of this type should be in the range of 10^{-2} to 10^{-3}.

From the study performed to evaluate the use of containment purge valves, we found the probability of long-term (recirculation) ECCS failure, given a loss of containment integrity, to be in the range of (2.5×10^{-1}) to 10^{-2}. For a PWR-3 accident, the frequency of a LOCA with ECCS cooling and loss of containment integrity using conservative values is $(10^{-3})(10^{-2})$/RY or 10^{-5}/RY. For a PWR-8 accident, the frequency of a LOCA with loss of ECCS cooling and loss of containment integrity using conservative values is $(10^{-5})(2.5 \times 10^{-1})$/RY or 2.5×10^{-6}/RY.

Consequence Estimate

The source term for the case of containment failure with effective long-term core cooling is that of a WASH-1400[16] PWR-8 event, LOCA with effective ECCS and loss of containment integrity from containment isolation failure. Without effective long-term core cooling following containment failure, the source term is that of a PWR-3 event, early containment failure and depressurization followed by ECCS failure in the recirculation mode. Consequences for PWR-3 and PWR-8 release categories are expressed in man-rem. The total whole-body man-rem dose is obtained by using the CRAC Code[64] for a particular release category. The calculations assume a uniform population density of 340 people per square mile (which is average for U.S. domestic sites) and a typical (midwest plain) meteorology.

For a PWR-3 event, $D = 5.4 \times 10^6$ man-rem
For a PWR-8 event, $D = 7.5 \times 10^4$ man-rem.

Cost Estimate

In approximating the cost, the following must be considered: (1) the cost to complete the generic issue, and (2) assuming an error is detected, the cost to correct the error at the operating ice condenser plants. There are 10 ice condenser plants, 4 currently operating and 6 in the OL review stage.

Industry Cost: Assuming an error is detected, the solution would require either a major design change to the ice condenser or addition of a new redundant large-capacity containment heat removal system. In either event, large plant downtimes would be incurred at the 4 operating plants with ice condenser containments. The cost of replacement power is assumed to be $300,000/day. We have assumed a minimum effective downtime of 90 days for corrective actions.

The cost of redundant heat removal systems would be on the order of millions of dollars. We will assume the cost to be $10M (including the cost of licensing review). The cost to each of the 4 operating plants would be (90)($300,000) + $10M. Therefore, the total cost for all 4 operating plants is $148M. The cost to the 6 unlicensed plants with ice condenser containments would be $(6 x 10)M or $60M.

NRC Cost: The cost to complete the generic issue was estimated to be 3 man-years of NRC and INEL (CSB consultant) personnel time. The code work was scheduled to be done in 1983. Therefore, the NRC cost would be about an additional $300,000 to find a human error if present. The NRC cost to complete the evaluation of the issue for all 10 ice-condenser reactors would be about $30,000/reactor. This cost is negligible in comparison with industry cost.

Therefore, the total cost associated with the solution to this issue is $(148 + 60)M or $208M.

Value/Impact Assessment

It is assumed that the average life is 35 years for the 10 reactors (4 operating and 6 under construction).

(1) For a PWR-3 event, the public risk reduction is 4.7×10^3 man-rem. Therefore,

$$S = \frac{4.7 \times 10^3 \text{ man-rem}}{\$208\text{M}}$$

$$= 23 \text{ man-rem/\$M}.$$

(2) For a PWR-8 event, the public risk reduction is 2.6×10^2 man-rem. Therefore,

$$S = \frac{2.6 \times 10^2 \text{ man-rem}}{\$208\text{M}}$$

$$= 1.2 \text{ man-rem/\$M}.$$

Uncertainties

The uncertainty of each of the constituents in the value/impact score equation is about a half order of magnitude. Therefore, the uncertainty of the calculations should also be about a half order of magnitude to an order of magnitude. Even if the value/impact score were an order of magnitude or more greater, it would not be large by comparison to scores for recognized high risk events ($S \geq 10^3$).

Additional Considerations

Work on developing an independent analytical model for ice condenser containment analysis has been pursued for about the last 5 years. The containment features necessary for the analysis of conventional dry containments have been checked out and accepted. The ice condenser model is operational and checkout is partially complete. CSB expects to have an accepted code with users manual by March 1982. Costs to date have probably exceeded $1M. Costs to complete the code should be small in comparison to the funds already expended.

In addition, the person utilized by INEL in the CONTEMPT-4 development and checkout effort will be needed to conduct the continuing code maintenance and CSB technical assistance functions. The CONTEMPT Codes are the CSB licensing evaluation tools and must be maintained. Past experience has shown that termination of code work or alterations in the scope of the INEL contract has resulted in the individual assigned to the CONTEMPT code leaving INEL and thereby causing a severe disruption in CSB consultant capability.

CONCLUSION

The low value/impact score arrived at above indicates that this issue should be dropped as a generic issue. However, if significant errors exist in the computer code, the containment could fail during an accident and one level of the "defense-in-depth" protection could be lost. Furthermore, the maintenance of consultant capability at INEL is essential to performing the CSB licensing reviews. Based on the above analysis, it was recommended that the CONTEMPT-4 Code development be completed with a medium priority. Any future application analysis performed by INEL or CSB should be addressed to individual plant reviews and charged as case work.

MOD 4 and MOD 5 to the COMPEMPT 4 User's Manual were published as NUREG/CR-3716[649] and NUREG/CR-4001,[650] respectively. No change to the SRP[11] was required since it already included a provision for the staff to perform confirmatory analyses.[648] Thus, this issue was RESOLVED and no new requirements were established.

REFERENCES

3. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," U.S. Nuclear Regulatory Commission, June 1978.

11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.

16. WASH-1400 (NUREG-75/014), "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, October 1975.

17. NUREG/CR-0255, "CONTEMPT-LT/028: A Computer Code for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident," U.S. Nuclear Regulatory Commission, March 1979.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

167. WCAP-8555, "LOTIC: Long-Term Ice Condenser Containment Code," Westinghouse Electric Corporation, April 1976.

648. Memorandum for T. Speis from H. Denton, "Closeout of Generic Issue B-54 'Ice Condenser Containments,'" October 22, 1984.

649. NUREG/CR-3716, "CONTEMPT 4/MOD 4," U.S. Nuclear Regulatory Commission, March 1984.

650. NUREG/CR-4001, "CONTEMPT 4/MOD 5," U.S. Nuclear Regulatory Commission, September 1984.

ITEM B-60: LOOSE PARTS MONITORING SYSTEMS

DESCRIPTION

The presence of a loose (i e., disengaged and/or drifting) object in the primary coolant system can be indicative of degraded reactor safety resulting from failure or weakening of a safety-related component. A loose part, whether it be from a failed or weakened component or from an item inadvertently left in the primary system during construction, refueling, or maintenance, can contribute to component damage and material wear by frequent impacting with other parts in the system. A loose part can pose a serious threat of partial flow blockage with attendant departure from nucleate boiling (DNB) which in turn could result in failure of fuel cladding. In addition, a loose part increases the potential for control rod jamming and for accumulation of increased levels of radioactive crud in the primary system.

The primary purpose of the loose part detection program is the early detection of loose metallic parts in the primary system. Early detection can provide the time required to avoid or mitigate safety-related damage to, or malfunction of, primary system components.

Applicants for construction permits and operating licenses are required to commit to a loose-part detection program. The NRC has developed hardware criteria and programmatic (operational) criteria for loose-parts detection programs. These criteria are contained in Regulatory Guide 1.133[146] which was issued in May 1981 after consideration of comments received from the industry.

Loose-parts detection programs are also in effect at most PWR operating facilities. For those programs which are generally consistent with the criteria contained in the proposed Regulatory Guide, operating experience has been very good.

The purpose of this NUREG-0471[3] task is to resolve any outstanding issues related to implementation of the Regulatory Guide, including the development of staff positions and guidance with respect to upgrading loose parts detection systems at operating facilities.

CONCLUSION

All CPs and OLs reviewed after January 1, 1978 were required to meet the provisions of the existing Regulatory Guide 1.133,[146] Revision 1. In addition, Regulatory Guide 1.133[146] recommended that owners of reactors licensed to operate prior to January 1, 1978 review their systems to determine if they were in compliance. Thus, this issue was RESOLVED and no new requirements were established.[670]

REFERENCES

3. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," U.S. Nuclear Regulatory Commission, June 1978.

146. Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," U.S. Nuclear Regulatory Commission, September 1977.

670. Memorandum for H. Denton from R. Mattson, "Generic Issue B-60, Loose Parts Monitoring Systems for Operating Reactors (TACS 52325)," January 10, 1984.

ITEM B-65: IODINE SPIKING

DESCRIPTION

Historical Background

This NUREG-0471[3] task is to develop and confirm a model for the iodine spiking phenomenon, in which the iodine concentration in the reactor coolant rises to many times its equilibrium concentration level (peak concentration) followed by a decay back to a level below the peak concentration. Procurement of data from operating plants and the development of a fuel release model for predicting the magnitude of the spikes will provide an understanding of this phenomenon which is not presently available. Improved knowledge of this topic would establish a better basis for accident calculations and could be used as a basis for establishing new reactor coolant activity limits.

Safety Significance

The calculated radiological consequences for some postulated design basis accidents are highly dependent upon the magnitude of the iodine spike assumed in the dose calculation model. These calculations are made with conservative assumptions, incorporating an iodine spiking factor which is based on a limited sample of plant data, and are in turn used to establish allowable coolant activity limits in the TS governing plant operations. However, the iodine spiking is a significant effect in only non-core melt accident consequences, which are not major contributors to nuclear plant risk.

PRIORITY DETERMINATION

A technical analysis of the proposed resolution of this issue was performed by PNL.[64] The resolution of this issue would apply to all operating and planned LWRs.

Frequency/Consequence Estimate

For converting thyroid exposure to equivalent whole body exposure, PNL derived a PWR expected public risk of 0.0143 man-rem/RY for a non-core melt SGTR and a coincident iodine spike using: (a) the PWR SGTR Task Force estimates for the probability of non-core melt SGTR events (1.3×10^{-3}/RY) and the amount of radioiodine (I-131) released (53,600 Ci/event); (b) the Prairie Island 1 conversion factor for translating curies of I-131 released to thyroid exposure; and (c) the conversion factor derived in the prioritization of Item III.A.1.3, "Maintain Supplies of Thyroid Blocking Agent." Using the ratioing technique described in NUREG/CR-2800[64] and a BWR small break LOCA frequency of 1.4×10^{-3}/RY, a BWR expected public risk due to a small break LOCA with a coincident iodine spike of 0.0185 man-rem/RY was derived.

Peak iodine concentration levels were estimated by AEB based on the average measured PWR and BWR coolant activity levels and an average peaking factor of 500, which was derived from the small population of data available on the iodine spiking phenomena. The peak primary coolant activity levels derived in this manner were estimated to be 60 μCi/gm and 4 μCi/gm for PWRs and BWRs, respectively, and represent the base case average peak iodine concentrations before resolution of this issue.

Dose calculations used by the STGR Task Force were performed using an assumed coolant iodine activity level increase by a factor of 20 and a maximum allowed primary coolant iodine concentration of 1.0 μCi/gm for PWRs and 0.2 μCi/gm for BWRs, or an allowable primary coolant peak iodine concentration of 20 μCi/gm and 4 μCi/gm for PWRs and BWRs, respectively. It was assumed that new coolant activity limits established after the iodine spiking phenomena was better understood and would not permit allowable peak iodine concentrations greater than those derived above. Thus, the above values are assumed to represent the adjusted case peak allowable coolant activity concentrations after resolution of this issue.

The thyroid dose was converted into a risk-equivalent whole-body dose using the assumptions that: (1) health effects from thyroid dose are 95% curable with no long-term effects, and (2) whole-body dose is given five times the weighting of thyroid dose (consistent with NRC protective action guides).

The post-implementation or adjusted public risk was determined by multiplying the pre-implementation or base case public risk by the ratio of the post-implementation reactor primary peak iodine concentration level to the pre-implementation average primary peak iodine concentration. As a result, the adjusted case public risk of 0.00477 man-rem/RY and 0.0185 man-rem/RY was calculated for PWRs and BWRs, respectively.

The change in public risk which might be realized by completion of this issue was determined by subtracting adjusted public risk from the base case public risk. The change in public risk was thus calculated to be 0.00953 man-rem/RY and 0 man-rem/RY by multiplying the above changes in public risk by the respective number of reactors and their average remaining lifetime (i.e., PWRs - 90 reactors and 28.8 years; BWRs - 44 reactors and 27.4 years) and adding the products. Total public risk reduction was estimated to be 25 man-rem for completion of this issue.

Since this iodine spiking issue does not significantly affect core-melt accident consequences, resolution of the issue would not result in a core-melt frequency change.

Cost Estimate

From the currently available data, it was judged that the 4-hour sampling interval following a transient, which is currently proposed in LCOs, would probably miss some spiking peaks. A change to a 2-hour interval was thus assumed to provide adequate information for peak activity determination. The total sampling period following a major power transient was estimated to be 33 hours. At a sampling interval of 2 hours, rather than 4 hours, it was estimated that 8 additional samples would be required following each major transient. A survey of the available iodine spiking data resulted in an estimated frequency of iodine spiking events of 0.52/RY and 0.14/RY for PWRs and BWRs, respectively.

Industry Cost: It was assumed that the costs to industry are due to the increased frequency of iodine sampling after each transient. No new equipment for sampling and analysis was assumed to be required. However, some minor modification of the sampling systems was assumed to be required at operating plants to accommodate the increased sampling frequency.

At the 71 operating plants, 4 man-weeks of labor were assumed to upgrade the sampling and analysis capability to accommodate the shorter sampling interval. At a cost of $2,270/man-week, a total industry implementation cost of $645,000 was calculated.

Increased industry operating costs were estimated using the 8 estimated additional samples per major transient, the above estimated iodine spike frequencies for PWRs and BWRs, the respective number of reactors and their average remaining life, and an estimated 2 man-hours to obtain and analyze a reactor coolant sample. A total industry operating cost of $1.38M was calculated. Therefore, the total industry cost associated with this issue was estimated to be $2M.

NRC Cost: Efforts required by the NRC to develop and confirm a model for the iodine-spiking phenomenon could be significant because little is known about the physics associated with the phenomenon. Two staff-years of NRC effort were estimated for the development of new requirements. Contractor support of the development of new requirements was estimated to be $300,000. At a cost of $100,000/man-year for NRC personnel, a total NRC cost of $0.5M for resolution of the issue and development of new requirements was estimated.

It was assumed that NRC staff time would be expended in the review of increased sampling requirements and the resulting information during the lifetime of the plants. It was estimated that 0.1 man-week/RY would be required to monitor the new sampling requirements and plant results at a total NRC cost of $860,000. Thus, the total NRC cost is estimated to be about $1.4M.

Value/Impact Assessment

Based on a public risk reduction of 25 man-rem, the value/impact score is given by:

$$S = \frac{25 \text{ man-rem}}{\$(2 + 1.4)\text{M}}$$

$$= 7.4 \text{ man-rem/\$M}$$

Uncertainties

Uncertainty in cost was found to be small, about ± 50%. Uncertainty in the public risk reduction estimate ranged from about plus 2 orders of magnitude on the upper bound to about minus 1 order of magnitude on the lower bound.

Other Considerations

It was assumed that all the labor associated with obtaining and analyzing additional record coolant samples would, of necessity, be expended in moderate radiation fields. In addition, one-fourth of the labor estimated for modification of the sampling systems at operating plants was assumed to occur in a moderate radiation field. Assuming a field of 25 millirem/hr a total increased ORE of 370 man-rem was estimated.

CONCLUSION

The total public risk reduction calculated for this issue is insignificant. Furthermore, the value/impact ratio is poor. The estimated increase in ORE due to the assumed resolution of the iodine spiking issue is large in comparison to the estimated public risk reduction, which would also be an incentive for a drop priority assignment. Uncertainty, although high for the public risk reduction estimate, would only support a remote possibility that the issue could warrant as high as a medium-priority assignment. Therefore, based upon the above considerations, we recommend that this issue be assigned a DROP priority.

REFERENCES

3. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," U.S. Nuclear Regulatory Commission, June 1978.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

ISSUE 12: BWR JET PUMP INTEGRITY

DESCRIPTION

Historical Background

A memorandum[22] from AEOD to NRR dated May 23, 1980 drew attention to the generic issue of BWR jet pump integrity. The concern that motivated the AEOD memo was a February 1980 jet pump failure at Dresden Unit 3, together with previous jet pump integrity-related problems at Dresden and Quad Cities. The Dresden failure was caused by progressive stress corrosion cracking of the pump's hold-down beam. The unit was shut down and the failed beam was replaced, along with six other beams for which indications of cracking were found upon ultrasonic inspection. Information concerning an earlier (May 1979) jet pump beam failure at a foreign GE BWR came to NRC's attention after the Dresden 3 failure.

Safety Significance

In GE BWRs (except in the earliest plants), water recirculation within the reactor vessel during normal power operation is accomplished by a ring of 16 to 24 water-jet pumps. Failure of a pump is of concern not only because of each pump's contribution to proper distribution of water flow within the vessel during normal operation but also because the pumps are designed to assure maintenance of water level well up in the core region in the event of a LOCA. The jet pump inlet is located about two-thirds of the way up the core height. If pump failure should lead to damage further down in the pump's diffuser, a lower-level outlet path could be opened and could prevent reflooding of the core following a break in a recirculation line. A degraded jet pump could be more vulnerable to damage from stresses due to water hammer or LOCA loads, should they occur. Also, jet-pump damage could permit increased rate of coolant loss in a LOCA since, in a LOCA, the jet pump's nozzle area is the limiting area for flow.

Possible Solutions

The AEOD memo[22] includes recommendations for generic corrective and evaluative actions. They include: (1) scheduled replacement of all hold-down beams with structures of improved design; (2) evaluation of the potential for water-hammer-type loads, the magnitude of such potential loads, and their impact on jet-pump integrity; and (3) evaluation of the potential for damaging vibration and fatigue failure during initial LPCI injection and subsequent long-term cooling modes. Further discussion of the AEOD recommendations appears in a responding memo[51] from NRR, dated July 11, 1980, in which phased replacement of the hold-down beams was discussed.

Currently, operating plants are monitoring jet pump performance in accordance with an IE Bulletin No. 80-07[52] issue[illegible] 1980 and information supplied in GE SIL No. 330.[53] GE has meanwhile de[illegible]ed and prototype tested improved hold-down beam bars. GE is now [illegible] to produce and sell the improved beams to replace the existing [illegible]s work is being reviewed by OIE and

Materials Engineering Branch (MTEB) of NRR. Definitive plans for any further steps for operating plants and for plants well along in construction remain to be formulated. Possible technical specification changes to include early indication of jet pump degradation remain to be addressed.

Possible means to gain early indication of hold-down beam damage include (a) monitoring the ratio of jet-pump driven flow to driving flow and (b) ultrasonic inspection of the beams for incipient cracks (at refueling--typically at about 18-month intervals). The efficacy of jet-pump performance monitoring depends on empirical and analytical indications that the breaking of a hold-down beam is preceded by a period during which a severely cracked beam allows some displacement of the jet nozzle thereby impairing the jet pump's performance. According to a GE estimate, this warning time would be about 7 to 13 days for the BWR/3 design; however, for BWR/4 plants, only about 1/3 to 2 days are expected to be available and the indication would be less clearly discernible. The value of ultrasonic inspection at refueling is based on the slow, progressive nature of the stress-corrosion cracking. GE estimates that small cracks can begin to appear several years after start of operation (about 4½ years for the BWR/3 plants and over 10 years for the newer BWR/4 plants), and that it takes about 1½ additional years for the cracks to propagate to failure.

PRIORITY DETERMINATION

Frequency Estimate

In some 100 RY (about 2000 pump-years) of BWR operation in the U.S. to date, there has been one hold-down beam failure (at Dresden 3). In the 14 months after the Dresden 3 failure, 19 additional beams were replaced because of crack indications in ultrasonic inspection. Eighteen of these indications occurred in 6 of the 8 BWR/3 reactors inspected, and one in one of the 12 BWR/4 reactors inspected. (Partial information on overseas reactors indicates one failure in one BWR/3 as well as some cracked beams detected in inspection.) Because of the nature of the problem--progressive stress-corrosion cracking--an increase in the rate of incipient failures with the hold-down structures now in the plants can be anticipated as their exposure time increases, unless corrective action is taken. Because cracks are detectable and plants have been alerted to the problem, there is reasonable prospect of corrective action before gross failure of the hold-down beams. However, after the one-time requirement imposed by IE Bulletin No. 80-07,[52] further periodic ultrasonic inspection of beams is not now required and the presence of at least one beam with some degree of cracking in a BWR/3 reactor at any time is very likely (probability = 1). The newer BWR/4 reactors appear to have a lower probability now (perhaps 0.1) but that probability may well increase towards the BWR/3 level over the next several years as the plants accumulate operating time. Accordingly, a probability of 1 for the presence of a crack in at least one beam appears to be a reasonable basis for a bounding calculation.

Jet-pump hold-down failure could lead to core damage by two types of mechanisms: LOCA aggravation and flow maldistribution. Frequency estimates for each of these mechanisms are as follows:

LOCA aggravation: A recirculation line break in addition to jet-pump damage could prevent core reflooding. The two required events may not be independent; stresses resulting from a line-break LOCA could damage a weakened jet pump. However, for major core damage to occur, the jet-pump damage must be severe enough to permit a coolant loss rate greater than the ECCS can replace, by opening a large flow area at a level well down in the core. For lack of better information, a conditional probability of 1 is conservatively assigned to LOCA aggravation (by a large enough hole being opened), given recirculation line break and given jet-pump hold-down breakage. Again, it is conservatively assumed that without periodic ultrasonic inspection and cautionary replacement of beams with incipient cracks, there is likely to be present a crack large enough to result in beam failure during a LOCA. As discussed, the probability of a cracked beam is 1. Thus, the contingent probability of LOCA aggravation by jet-pump hold-down beam failure, given recirculation-line break, is (1)(1) or 1. Based on WASH-1400[16] (Sec. III-6.4) and NUREG/CR-1659[54] (Vol. 4, pp 4-22), the estimated frequency of a recirculation line break is 10^{-4}/RY.

For core damage to occur, the core spray must also fail. If, on the basis of WASH-1400[16] (p. II-5), one takes the probability of such failure as 10^{-3} (assuming core spray failure to be independent of jet-pump hold-down and diffuser failures), the estimated frequency (F) of core damage due to LOCA aggravation by jet-pump beam failure is given by $F = (1.0)(10^{-4})(10^{-3})$/RY or 10^{-7}/RY.

A flow monitoring and ultrasonic inspection program should provide a substantial reduction in this probability because, as discussed above, crack growth is slow and a hold-down beam weakened by a substantial crack is believed necessary for LOCA aggravation. At least a ten-fold improvement should be typically obtainable.

Flow Maldistribution: Loose broken-off parts could obstruct flow in part of the core substantially aggravating maldistribution of flow due to loss of a jet pump. However, detection and correction before substantial core damage could occur can be relied on with a high probability. It is believed that the overall risk from the BWR jet-pump problem can be estimated to a fair approximation in terms of the LOCA aggravation mechanism alone.

Consequence Estimate

The estimated consequences are the same as those for a WASH-1400[16] BWR-2 release category. In this accident category, decay-heat-removal systems are assumed to fail (i.e., LPCI and LPCS). As a result, containment fails by overpressure, core melting occurs, and release of radioactive materials takes place over about 3 hours without significant retention of fission products. The choice of the BWR-2 category reflects prompt failure of decay heat removal after a transient occurs while the reactor is at power. Consequences for a BWR-2 release category are expressed in man-rem. The total whole-body man-rem dose is obtained by using the CRAC Code for the particular release category. The calculations assume a uniform population density of 340 people per square mile (which is average for U.S. domestic cities) and a typical (midwest plain) meteorology. For a BWR-2 accident, consequences are 7.1×10^6 man-rem.

Cost Estimate

Industry Cost: Based on estimates informally received from GE, 20 operating plants would each require the following typical costs for change-over of all hold-down beams:

Hardware and labor to install	$ 100,000
Plant outage (cost of replacement power for 10 days of additional outage time at refueling)	$3,000,000
Total cost per typical BWR Plant =	$3,100,000

For each operating reactor, the costs of complete beam change-out would be reduced, to an extent that would be plant specific, if those costs were adjusted for the averted replacement cost of failed or damaged beams. For plants of the older BWR/3 type, it is quite likely that most of the original beams would eventually show damage and require replacement. For BWR/4 reactors, corrective replacement of damaged beams may be lower in proportion and mostly further in the future. Damaged beams must be replaced for operational as well as safety reasons.

No significant industry costs are expected to be associated with the use of improved hold-down beams in future plants. For plants nearing completion in which hold-down beams would be replaced, a cost of $100,000 per plant may be assumed.

The cost of ultrasonic inspection is estimated to be $10,000/reactor for a 1.5-year inspection interval, on the assumption that it can be accommodated during a refueling outage without prolonging the outage. The cost of flow monitoring is estimated to be small in comparison. Thus, monitoring costs may be taken to be about $0.01M/RY. (This includes $0.007M/RY for inspections plus $0.003M/RY for flow monitoring.) The equivalent one-time cost at "present worth" for a remaining reactor life of 20 to 30 years may be taken as approximately $0.1M/reactor.

Replacement of suspect beams during refueling is estimated to cost about $150,000/beam, on the assumption of a 1/2-day lengthening of a refueling shutdown per beam replaced. To replace a beam in a shutdown forced by beam breakage during operation or by flow-monitoring evidence of hold-down beam separation would cost typically some $2.1M for replacement power (at $0.3M/day) during a 1-week plant outage. This makes cautionary replacement of beams with ultrasonic indication of incipient cracking cost-effective, since the likelihood of failure before the next refueling would have a probability higher than 0.07 (i.e., higher than $0.15M/$2.1M).

NRC Cost: NRC costs are negligible in comparison with industry costs and include:

(1) Monitoring and ultrasonic inspection costs for 20 operating plants are $(20)(0.1)M or $2M.

(2) Beam replacement costs for 20 operating plants are \$(20)(3.1)M or \$62M.

(3) Beam replacement costs for plants nearly completed are \$0.1M/reactor.

Value/Impact Assessment

The risk reduction per reactor based on a 30-year operating life is $(7.1 \times 10^6)(10^{-7})(30)$ man-rem/reactor or 21 man-rem/reactor. The value/impact scores are as follows:

(1) For operating plants, based on imposition of jet-pump flow ratio monitoring during operation and ultrasonic inspection at each refueling (before adjustment of the score for the cost-effectiveness of the monitoring program on the basis of plant economics):

$$S = \frac{(7.1 \times 10^6)(10^{-7})(20)(30)\ \text{man-rem}}{\$2\text{M}}$$

$$= 210\ \text{man-rem/\$M}.$$

If this score were adjusted for the previously noted cost advantage of cautionary over emergency beam replacement, the score would become negative, indicating that the safer course is also the more economical in this case.

(2) For operating plants, based on complete replacement of existing beams with beams of improved design, before adjustment for averted cost of replacing cracked beams:

$$S = \frac{(7.1 \times 10^6)(10^{-7})(20)(30)\ \text{man-rem}}{\$62\text{M}}$$

$$= 7\ \text{man-rem/\$M}.$$

This value/impact score would become more favorable if it were adjusted for the averted cost of replacing cracked beams. The magnitude of this effect can vary widely (from insignificant to possibly dominant) according to plant specifics.

(3) For nearly completed plants, the value/impact score is as follows:

$$S = \frac{(7.1 \times 10^6)(10^{-7})(N)(40)\ \text{man-rem}}{\$(0.1 \times N)\text{M}}$$

$$= 280\ \text{man-rem/\$M}.$$

This score reflects an assumption that the beam changeover would not be on the critical path in the schedule of remaining activities leading to initial power operation. Therefore, only hardware and labor costs are included. Adjustment for the averted cost of replacing cracked beams could well make this score negative, indicating that changeover of the beams would be beneficial from the standpoint of plant economics as well as safety.

(4) For future plants, the added cost for beams of improved design is negligible. Thus, the value/impact score obtained for nearly-completed plants makes the installation of beams of improved design a favorable option for future plants.

Uncertainties

In addition to the usual wide uncertainties about estimates of accident frequencies and consequences, this issue is highly sensitive to uncertainties in estimates of two cost elements:

(1) A high frequency of corrective or cautionary replacement of failed or damaged beams could wipe out the costs saved by not making a complete changeover to improve beams.

(2) The cost of replacement power is the dominant element in beam replacement costs on operating plants. These can vary widely among plants. Also, if beam replacement can be scheduled at a period of low power demand, replacement power costs could greatly decrease. The priority score could thus become selectively much more favorable for some specific plants at some specific changeout times.

Other Considerations

(1) The refloodable core, which depends on jet-pump integrity, is an important layer of protection against core-melt as part of the defense-in-depth concept embodied in BWR designs.

(2) This issue has attained a proposed technical resolution.

(3) As noted in the preceding discussion, the probability of beam cracking can vary widely according to plant specifics, notably design and age. The benefit-cost relations in possible beam changeout are highly variable according to case specifics.

(4) The estimated economic impact of implementing an available resolution that involves use of improved-design beams increases sharply as one proceeds from a pre-construction stage to construction and to operation.

(5) The available monitoring program is cost-effective.

CONCLUSION

After considering the value/impact scores calculated for the various categories of new and existing plants, together with the other considerations stated above, it was recommended that the further steps of establishing regulatory requirements and implementation plans appropriate to the various specific situations be pursued. As a result, this issue was given a medium priority ranking.

In response to IE Bulletin No. 80-07,[52] 7 plants found no evidence of cracking or unusual wear; however, 3 of these plants replaced their hold-down beams with new improved-design beams. Thirteen plants found damaged beams; 11 of these replaced their beams with either new beams of the same design, but with reduced preload, or with beams of a new design, with an improved heat treatment. The other 2 plants replaced all beams with beams of the new design.

Those plants that did not replace all hold-down beams with the new design beams have continued to do daily flow rate surveillance, as required by IE Bulletin No. 80-07,[52] and to voluntarily perform ISI of the beams at refueling outages. Plants under licensing review have either changed the hold-down beams with those of the improved design or have committed to a surveillance and ISI program.

The staff reviewed the voluntary actions taken by OLs and CPs and concluded that the issue of possible failure of BWR jet pump hold-down beams has been adequately addressed.[666] Thus, this issue was RESOLVED and no new requirements were established.

REFERENCES

16. WASH-1400 (NUREG-75/014), "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, October 1975.

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51. Memorandum for C. Michelson from H. Denton, "BWR Jet Pump Integrity," July 11, 1980.

52. IE Bulletin No. 80-07, "BWR Jet Pump Assembly Failure," U.S. Nuclear Regulatory Commission, April 4, 1980.

53. SIL No. 330, "Jet Pump Beam Cracks," General Electric Company, BWR Product Service, June 9, 1980.

54. NUREG/CR-1659, "Reactor Safety Study Methodology Application Program," U.S. Nuclear Regulatory Commission, 1981.

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ISSUE 22: INADVERTENT BORON DILUTION EVENTS

DESCRIPTION

Historical Background

Many PWRs have no positive means of detecting boron dilution during cold shutdown.[25] Some operations carried out during outages (e.g., steam generator decontamination) reduce the RCS volume, thus speeding up dilution. Boron dilution has taken place during such operations although, thus far, criticality has not occurred.[26] An analysis of this issue was provided in a DST memorandum.[108]

Safety Significance

There have been 25 reported instances of inadvertent boron dilution during maintenance and refueling.[109] Although none has yet occurred, the safety concern is the possibility of an inadvertent criticality. If the boron is sufficiently diluted and the reactor core is near beginning of cycle, it is possible to bring the reactor to criticality with all of the control rods inserted into the core. The only way to shut the core down again in such a circumstance would be to reborate the moderator, which could take considerable time. The events have occurred with sufficient frequency to raise the question whether, considering their possible consequences, the degree of protection is appropriate.

Possible Solution

All 43 operating PWRs are affected by this condition. The fix is to install instrumentation to detect the event and stop the dilution either automatically or, if the detection is sufficiently early, by alerting the operator.

PRIORITY DETERMINATION

Frequency Estimate

Boron dilution events during a shutdown or refueling have usually been caused either by human error or by failures of special, nonprocess equipment such as inflatable seals. Therefore, event frequencies cannot be easily calculated by fault tree analysis. Moreover, because no event has yet resulted in criticality, it is not possible to simply add up the number of events in operating history.

The fact that no inadvertent criticalities have happened in 337 PWR-years allows us to estimate an upper bound to the frequency. By assuming a Poisson distribution and using a 95% confidence level, the frequency of inadvertent criticalities is, at most, 9×10^{-3} event per PWR-year.

However, an upper limit is not sufficient to gauge the significance of boron dilution events; a "best estimate" (in some sense) is needed. The only information available is contained in the frequency of boron dilution events which have happened but which did not result in criticality. Most of these events can be considered "precursor" events to an actual inadvertent criticality.

The severity of a precursor event is defined here in terms of the shutdown margin remaining at the end of the event. That is, an event which was halted with 2% shutdown remaining is considered more severe than an event which was halted with 10% remaining shutdown margin.

Using the information in the NUREG/CR-2798,[110] a histogram shows that the number of events goes down as the severity increases. To estimate an expectation value for the number of critical events, a two-parameter exponential distribution was fitted to the data. Extrapolation of this distribution to the point of zero shutdown margin gives a value of 0.67 event in a time interval of (currently) 337 PWR-years. Thus, we expect the frequency of inadvertent criticalities to be on the order of 2×10^{-3} event per PWR-year.

This calculation, although rough, gives an answer that is reasonable. With 43 PWRs presently operating, we would expect an inadvertent criticality roughly every 11 years, if nothing were done.

However, this number does not take into account the effect of the neutron monitoring instrumentation. As a reactor core approaches criticality, neutron flux does not rise linearly. Instead, the reciprocal of the flux drops linearly as shutdown margin decreases. The net effect is that neutron flux rises slowly as the reactor core goes from 10% to 9% shutdown, but rises very dramatically as shutdown margin drops below 0.5%. None of the events tabulated in NUREG/CR-2798[110] came close enough to criticality for the neutron monitoring channels to trigger alarms. Thus, to realistically estimate the frequency of an event that continues in dilution to criticality, we must give some credit for the neutron flux channel alarms, which are usually set one-half to one decade above background.

Since the control rods are already fully inserted into the core in this event, the only actions which will prevent criticality are stopping the dilution or reborating the moderator. Both are done by the operator. Thus, the credit to be given for neutron flux alarms is governed by the reliance which can be placed on the operator. We will assume (based purely on judgment) that the operator will be able to correctly diagnose the problem and successfully prevent criticality 90% of the time. This drops the frequency of a criticality by one order of magnitude, to 2×10^{-4} event/RY. Of these, roughly one sixth will take place with the reactor head removed. Thus, the frequency of radioactivity-releasing events is 3×10^{-5}/RY.

Consequence Estimate

In the PWR case under consideration here, all rods are either already in the core or are disconnected from their drives. Either way, there is no scram reactivity available. Shutdown by emergency boration will take much more time than shutdown via scram. The important parameter is the peak level achieved by the core.

Once the core becomes critical, it will heat up with a positive period governed by the rate of dilution and by moderator temperature and Doppler feedback. Eventually the coolant may boil and the peak power level will be limited by void generation in the moderator. Preliminary calculations indicate that, assuming BOC parameters (worst case), a power level of about 3% of rated would be reached.[111] (These calculations are limited in their ability to model the multidimensional aspects of void feedback.)

A core power of 3% of rated is not likely to fail fuel that must withstand decay heat rates of this same order. The only likely consequence is the release of gap activity from any leak already present. If we make the standard assumption of users of the GALE codes that 0.16% of the fuel leaks, the total activity released to the coolant would be roughly 69,000 Ci. This is not enough activity to be significant unless the vessel head is removed. If the vessel head were not in place, about 10% of this activity, or 6,900 Ci, would escape from containment, based on analyses of dropped fuel assembly events. Consequences for this event are expressed in man-rem. The total whole-body man-rem dose is obtained by using the CRAC Code[64] for the particular release category. The calculations assume a uniform population density of 340 people per square mile (which is average for U.S. domestic sites) and a typical (midwest plain) meteorology. Therefore, the dose for such as event would be 700 man-rem.

For 43 PWR operating plants with an average remaining life of 30 years, the total risk reduction is $(43)(3 \times 10^{-5})(7 \times 10^{2})(30)$ man-rem or 27.1 man-rem.

Cost Estimate

Industry Cost: Since these events are caused by a wide spectrum of causes, it is not practical to reduce the frequency of boron dilution events other than by bringing the matter to the attention of plant operations personnel and having them upgrade their procedures (if and where appropriate). It has been proposed to install a microprocessor-based monitor on the source range neutron flux instrumentation. Such a monitor, if connected to a display panel such as the safety parameter display system (SPDS), could give earlier warning of loss of shutdown margin than is possible with the present instrumentation, and thus would reduce the probability of a boron dilution event leading to criticality.

We have evaluated[112] the cost of such a system. The results are:

Control grade instrument, alarm only	-- \$ 50,000
Safety grade instrument, alarm plus automatic initiation of emergency boration	-- \$ 300,000

To be conservative, we assumed that the cheapest hardware fix at a cost of \$50,000/plant would be used. Therefore, the total industry cost is \$(0.05)(43)M = \$2.2M.

NRC Cost: The cost to the NRC is estimated to be 2 staff-months plus 1 staff-week for each of the 43 operating PWRs. This corresponds to an NRC cost of \$84,000 which is small in comparison to the cost of industry.

Value/Impact Assessment

Based on a total public risk reduction of 27.1 man-rem, the value/impact score is given by:

$$S = \frac{27.1 \text{ man-rem}}{\$2.2M}$$

$$= 12 \text{ man-rem/\$M}$$

Uncertainties

The upper limit (95% confidence) on inadvertent criticality frequency without credit for neutron flux alarms was a factor of 5 over the "best" estimate. If we assume a symmetrical distribution and also assume a factor of 5 error in the credit for the neutron flux alarms and a factor of 3 error in the chance of the head being off the vessel, the estimated error in the frequency of radioactive release is plus or minus a factor of 8.

The release is expected to be on the order of 6,900 Ci, primarily noble gases. We will use an estimated error of a factor of 5, again based on judgment.

The uncertainty in the costs, which are dominated by the $50,000/plant, is at most a factor of 2.

CONCLUSION

Based on the low value/impact score and low risk reduction associated with an inadvertent criticality, DST concluded that boron dilution events do not constitute a significant risk to the public and recommended that the issue be dropped from further consideration. However, DSI disagreed with this evaluation and obtained permission from the NRR Director to pursue the issue further.

As a result of DSI's work, it was determined that the consequences of an unmitigated boron dilution event, although undesirable, are not severe enough to warrant backfit of additional protective features at operating plants. DSI recommended that DL issue a generic letter to OLs informing them of this result and pointing out that the event represents a breakdown in a licensee's ability to control its plant. DSI concluded that the criteria in SRP[11] Section 15.4.6 are adequate for plants currently undergoing license review.[693] Furthermore, because offsite consequences following the event are likely to be insignificant, DSI also recommended that SRP[11] Section 15.4.6 be considered for deregulation.[694] This recommendation is covered in Issue 104. Thus, this issue was RESOLVED and no new requirements were established.

REFERENCES

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109. Memorandum for T. Murley from R. Mattson, "Inadvertent Boron Dilution," September 15, 1981.

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111. Letter to R. T. Curtis (NRC) from N. S. DeMuth (LANL), "Analyses of Unmitigated Boron Dilution Events," November 18, 1981.

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693. Memorandum for H. Denton from R. Bernero, "Resolution of Generic Issue No. 22, Inadvertent Boron Dilution Events (BDES)," September 17, 1984.

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ISSUE 49: INTERLOCKS AND LCOs FOR CLASS 1E TIE BREAKERS

DESCRIPTION

Historical Background

In an AEOD memorandum[638] to NRR, it was concluded that the design of the Point Beach Nuclear Plant, Units 1 and 2, under certain conditions, allowed manual interconnection of redundant electrical load groups, thereby paralleling their power sources. AEOD noted in its memorandum that it took the plant operators approximately five weeks to discover that the electrical distribution system line-up was not in the proper configuration. This suggested a generic concern regarding the adequacy of procedural and administrative controls. In this instance, the lack of procedures to include the monitoring of the status of the plant electrical distribution system during plant operation, through several shift changes, prevented the detection of the human error committed.

The incident referred to in the AEOD memorandum[638] was discovered (and corrected) at Point Beach Unit 2 on June 9, 1980.[639] It involved operation at 100% power with one of the 4160 V Class 1E redundant buses being supplied by the offsite power source via the other 4160 V redundant bus and its tie breaker.

In responding to AEOD, NRR identified[507] an additional complementary concern: interlocks are not provided to prevent the tie breaker from being closed when both normal feed breakers to the Class 1E buses are closed. Interlocks are not provided between the emergency diesel-generator output breakers and the tie breaker. Such interlocks should also be provided to prevent out-of-synchronization interconnection of the diesel generator and the offsite power source.

Safety Significance

GDC-17 requires that the onsite source and distribution systems have sufficient independence and redundancy to perform their safety function assuming a single failure. Operating the plant in the reported Point Beach configuration violates the independence requirement of being able to accommodate a single failure.

With features of breaker operation such as those at Point Beach the following problems potentially impair plant safety: (a) a failure of the tie breaker to open on loss of voltage would prevent both emergency diesel-generators from automatically supplying power to their respective buses (single failure); (b) the tie breaker is capable of being closed when the offsite source breaker is closed on one bus and the respective diesel-generator breaker is closed on the other bus (paralleling two divisions, one with offsite and the other with emergency sources); and (c) the tie breaker is capable of being closed when both 4160 V Class 1E buses are being supplied by their respective diesel generators (paralleling redundant emergency sources). This is contrary to the requirement of Regulatory Guide 1.6, which states "If means exist for manually connecting redundant load groups together, at least one interlock should be provided to prevent an operator error that would parallel their standby power sources."

Possible Solution

For purposes of this analysis, the assumed resolution of this safety issue is to require that the NRC issue an IE circular requiring all holders of operating licenses to review the design and operational features of all Class 1E bus tie breakers.

If only one tie breaker exists between redundant Class 1E buses, then the licensee should promptly take, as a minimum, the following actions, via procedural requirements (these are taken as the proposed resolution for affected plants): (a) use a bus tie breaker only during shutdown when it is absolutely necessary; (b) physically disengage each tie breaker and rack out (withdraw) following each usage; (c) "red tag" the tie breaker enclosure for the breaker to be kept open; and (d) incorporate QA procedures to reconfirm that all tie breakers are racked out and "red tagged" prior to each plant startup.

The present licensing practice as stated in SRP[11] Section 8.3.1, III.2.B, requires physically separated tie breakers in series between redundant Class 1E buses. In addition, the STS for new plants require tie breakers between redundant buses to be open as a condition of operability of the redundant Class 1E electrical distribution system. Therefore, this issue only affects operating BWRs and PWRs. A cursory review of AC one-line diagrams for 22 plants indicates that ten of these have single tie breakers between redundant buses.[64] Therefore, it is assumed that this issue affects 10/22 or 45% of all backfit LWRs, i.e., 11 backfit BWRs and 21 backfit PWRs.

PRIORITY DETERMINATION

Frequency Estimate

There has been one documented tie breaker failure to date.[64] Based on this experience (in some 1,000 RY of operation to date), one can estimate a tie breaker failure frequency of about 1×10^{-3}/RY.

The probability of an emergency diesel-generator failing on demand to supply power to Class 1E buses is conditional on the tie breaker between these buses being in a failed position during a loss of offsite power. Failure of emergency diesel power must persist for some time, typically 4 hours (based on the draft NUREG-1032), to lead to a station-blackout core-damage sequence.

The frequency of loss of offsite power (LOOP) of 4 hr. or longer duration is taken as 9×10^{-3}/RY, based on historic experience documented in the draft NUREG/CR-3992. With a mean time to discovery and repair of 5 weeks (i.e., 0.1 yr.) for the failed circuit breaker (based on the Point Beach history), the indicated frequency of station blackout with core damage becomes $(9 \times 10^{-3})(0.1)(1 \times 10^{-3}) = 9 \times 10^{-7}$/RY. If there is a 0.1 probability that the incorrect tie breaker position is not discovered and corrected within 4 hours after onset of LOOP, the adjusted core-damage frequency becomes $(0.1)(9 \times 10^{-7})$/RY or 9×10^{-8}/RY. With 32 reactors affected, the total core damage frequency for all affected reactors is $(32)(9 \times 10^{-8})$/yr. or 3×10^{-6}/yr.

Consequence Estimate

In view of the low estimated severe core damage frequency, no consequence estimates were made. Had such estimates been made, they would not have exceeded about 6×10^6 man-rem (about 5×10^6 man-rem for a PWR, based on a PWR-1 release and 7×10^6 man-rem for a BWR, based on a BWR-2 release).

Cost Estimate

PNL estimates[64] $330,000 for industry costs and $32,000 for NRC costs as the total for all plants. This is based on 42 man-hours of industry labor for each of 32 operating plants performing design review and taking corrective action and 6 man-hours/plant doing design review but not required to take corrective action. The combined industry and NRC cost for plants involving corrective action is approximately $10,000/plant. We multiply this figure by 2, arriving at $20,000/plant, including allowance for the added costs of approval reviews of analyses, designs, and reports and quality assurance measures for corrective actions.

Value/Impact Assessment

In view of the low estimated severe core damage frequency, a bounding assessment based on the upper-bound consequence of 6×10^6 man-rem (and 30-yr. remaining plant life) would show a public risk reduction of (30) $\times 10^{-8}$)(6×10^6) = 16 man-rem/reactor. For the 32 affected reactors, the . l risk reduction would be (32)(16) man-rem or approximately 500 man-rem. The upper-bound value/impact score for plants that would involve corrective action is as follows:

$$S = \frac{16 \text{ man rem/reactor}}{\$0.02\text{M/reactor}}$$

$$= 800 \text{ man-rem/\$M}$$

Uncertainties

The bounding consequence estimates may well be an order of magnitude too high. Depending on containment features and accident particulars, containment failure may be delayed or not take place at all.

The accident frequency may have been underestimated to the extent that there may have been unreported closed-tie-breaker events. Not all plants have a requirement to report Point Beach type incidents. (A number of older plants do not.) Thus, the one reported closed-tie-breaker incident may not represent all actual such incidents.

The risk analysis is based on the assumption that if the incorrect tie-breaker position is discovered and corrected, then the plant is safe from core damage. However, some plants may have no interlocks to prevent closure of the diesel-generator breakers upon LOOP with the tie breaker closed. In such designs, the diesel-generators can suffer damage such that they could not be restarted after opening the tie breaker. For such plants, core damage becomes an order of magnitude more likely (since the estimated 0.1 probability of failure to achieve effective correction before core damage would not apply).

CONCLUSION

The low estimate for potential risk reduction and the associated value/impact score indicate that this issue should have a low priority. However, we have assigned a MEDIUM priority because of the possible existence of plants having features that exacerbate the risk from this issue by causing potential serious damage to the diesel-generators.

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639. Letter to J. Keppler (NRC) from C. Fay (Wisconsin Electric Power Company), "Docket No. 50-301, Point Beach Nuclear Plant Unit 2 Licensee Event Report No. 80-005/03L-0," June 27, 1980.

ISSUE 50: REACTOR VESSEL LEVEL INSTRUMENTATION IN BWRS

DESCRIPTION

Historical Background

In January 1982, AEOD published a report (AEOD/C201[322]) on safety concerns associated with reactor vessel level instrumentation in BWRs. The report was forwared to NRR for further action.

Safety Significance

BWRs use reactor level instrumentation to perform a number of functions including control functions, such as feedwater control, and protective functions, such as automatic scram and autostart of emergency core cooling systems. AEOD concluded that, depending on specific plant instrumentation configurations, there could be the potential for adverse interactions between the control systems and the protection systems. As an example, the interactions may lead to loss of reactor water level due to automatic termination of normal feedwater (control) and failure to automatically start the emergency feedwater source (protection).

Possible Solution

The AEOD report[322] made three recommendations which were believed to be necessary to resolve the safety concern. The recommendations were:

(1) Action should be implemented to assure that automatic and manual safety-related low-low level start and high pressure injection functions of HPCI and RCIC turbines are not prevented or delayed by the non-safety-related high level trip. For example, the control system of HPCI and RCIC turbines could be modified to provide a low-low level start signal which overrides the high-level trip signal.

(2) Action should be implemented to assure that protective functions are provided in spite of any adverse control system/protection system interaction in the narrow-range level instrumentation. For example, the protective functions provided by the narrow-range level sensors could also be provided by the wide-range level sensors (in employing the wide-range level instrumentation, the desired output signal quality in terms of sensitivity, resolution, accuracy, and repeatability must be considered to assure that the initiating signals achieve the required protection function.) This approach would be consistent with the concept of "alternate channels" as defined in paragraph 4.7.4.1 of IEEE 279-1971.[397]

(3 Control room operators should be trained to recognize spurious vessel level indications and procedures should be provided for corrective actions to mitigate the consequences of potential transients that may be caused by level instrumentation malfunctions. We believe that the BWR emergency procedure guidelines provide the best vehicle for the definition of appropriate corrective actions in the event of level instrumentation malfunctions.

NRR responded to the three AEOD recommendations by describing a set of ongoing actions.[323] The ongoing actions, although related to the concerns, did not specifically address the AEOD recommendations.

In response to the concerns of this issue, the BWR Owners Group commissioned S. Levy, Inc. to study the reactor water level systems. As a result of this study, SLI-8211[411] was prepared in July 1982 and submitted to the staff for review. This report identified the three basic areas that water level instrumentation could be improved:

(1) Temperature effects causing decalibration and flashing,
(2) Failures and malfunctions of mechanical level indication equipment,
(3) Break in an instrument line combined with a single failure.

CONCLUSION

Although the staff believed that improvements in all three areas would be prudent, only the first two were found necessary to satisfy TMI Action Plan Item II.F.2 and produce substantial improvement in the water level instrumentation.[695] The affected licensees voluntarily agreed to make the necessary improvements and plans for implementing them were requested from the licensees by DL in Generic Letter No. 84-23.[696] Licensee actions in response to this request will be tracked under MPA F-26. The third area of improvement will be addressed in Issue 101.[697] Thus, this issue was RESOLVED and no new requirements were established.

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411. SLI-8211, "Review of BWR Reactor Vessel Water Level Measurement Systems," S. Levy, Inc., July 1982.

695. Memorandum for T. Speis from H. Denton, "Closeout of Generic Issue 50, 'Reactor Vessel Level Instrumentation in BWRs,'" October 17, 1984.

696. NRC Letter to All Boiling Water Reactor (BWR) Licensees of Operating Reactors (Except LaCrosse, Big Rock Point, Humboldt Bay and Dresden-1), "Reactor Vessel Water Level Instrumentation in BWRs (Generic Letter No. 84-23)," October 26, 1984.

697. Memorandum for D. Eisenhut from R. Bernero, "Resolution of Generic Issue 50, Reactor Vessel Level Instrumentation in BWRs," September 6, 1984.

ISSUE 53: CONSEQUENCES OF A POSTULATED FLOW BLOCKAGE INCIDENT IN A BWR

DESCRIPTION

Historical Background

In response to a 1967 ACRS concern relative to the potential of melting and subsequent disintegration of a portion of a fuel assembly due to inlet orifice flow blockage, GE submitted NEDO-10174[380] in May 1970. As a result of a staff review of another topical report on new fuel design, the applicability of NEDO-10174[380] to the then new (8 x 8) fuel was questioned. In late 1977, GE submitted NEDO-10174, Revision 1.[288] This revision is still awaiting staff review.[381]

Safety Significance

In a BWR, each fuel bundle is surrounded by a channel box. (The bundle plus channel box is referred to as a fuel "assembly.") Coolant is metered to each fuel assembly by individual orifices in the fuel support castings. All core flow must come through these orifices. Even the water between the channel boxes (outside of the fuel assemblies) now comes from holes drilled in the fuel assembly nosepieces; it does not come directly from the lower plenum.

The safety concern is that, if an orifice became blocked with the reactor operating at power, its corresponding fuel assembly could be deprived of coolant flow, but would still be producing thermal energy and could be damaged or even become molten. Moreover, if a highly overheated assembly were to suddenly refill with water (from dislodging of the blockage at the orifice), there would be the possibility of a steam explosion with consequent pressure pulses on the primary system.

Possible Solutions

Some mitigating features already exist. The first is the effect of the void reactivity coefficient. Reduced flow and consequent increased voiding will reduce power in the blocked assembly and in the immediately surrounding assemblies. This effect tends to compensate for most partial blockages. Second, even if the orifice is completely closed, the blocked assembly is still open at the top and has holes drilled in the nosepiece down below. Thus, some flow would still be available. (Older fuel designs did not have holes in the nosepiece. However, these older designs also did not have finger springs to seal the lower end of the channel box to the nosepiece. Thus, these older designs have as much or more leakage flow than the new designs.)

NEDO-10174, Revision 1[288] contains a summary of the calculations of the core's response to an orifice blockage event assuming these two features. The report came to the following conclusions:

(1) The only mechanism capable of causing a major flow blockage is that induced by a foreign object.

(2) Fragmentation, crudding, or fuel swelling cannot cause major flow blockages.

(3) A fuel assembly is capable of withstanding very severe blockages before losing adequate cooling.

(4) For orifice blockages greater than 98%, fuel and cladding melt are expected to occur. However, this will not result in failure propagation to adjacent assemblies, local high pressure production or off-site doses in excess of a small fraction of 10 CFR 100 guidelines. For this worst-case event, no action is required of the reactor ECCS. However, the reactor must be scrammed by the main steam line radiation monitor.

(5) For orifice blockages between 95% and 98%, clad melting is expected, but fuel melting is not calculated to occur. For this case, the consequences are less severe than in (4) above.

(6) For orifice blockages between 79% and 95%, boiling transition and attendant cladding heatup are calculated to occur. No clad nor fuel melting is calculated. However, cladding failure is not precluded. The off-gas radiation monitor will provide an alarm to the reactor operator if fission product releases are significant.

(7) For orifice blockages less than 79%, nucleate boiling is maintained. Therefore, the fuel and cladding are unaffected.

If these conclusions are accepted, it is not clear what more needs to be done to prevent damage from blockage. PNL suggested (for prioritization purposes) adding more holes to the nosepieces of the fuel assemblies.[64] These holes would normally be closed off by internal flapper valves. If the orifice were blocked, these springless valves would open to allow more coolant into the assembly.

PRIORITY DETERMINATION

Frequency Estimate

Experience has shown that the presence of loose objects in the primary system of reactors is not a rare occurrence. Nevertheless, blockage of an orifice is expected to be rare:

- coolant velocities are relatively low in the lower plenum -- so much so that thermal stratification might occur if the RWCU system did not continuously draw some water out of the lower vessel head drain. Thus, it is difficult for a non-floating object to be levitated by coolant flow.

- Most materials which could be carried up to the orifices (e.g., the Browns Ferry rubber shoe cover and the Duane Arnold aluminum can) chemically decompose very rapidly at reactor coolant temperatures and pressures.

- most loose parts, whether of internal or external origin, are made of stainless steel, Zircaloy, or some other material which does not float. In BWRs, such objects tend to settle to the bottom of the lower plenum.

Orifice blockage events may or may not be discovered by inspection. However, since blockages change power level in an assembly, blockage would also affect power maps and spent fuel isotopics. Thus far, to the best of our knowledge, no blockage events have occurred.

Presently, about 370 BWR-years of experience have been accumulated. No events have been reported in this interval. Thus, to 95% confidence, the frequency of detectable blockage events is limited by:

$$F \leq -\frac{\ln(0.05)}{370\ RY}$$

$$F \leq 8.1 \times 10^{-3}/RY$$

In reality, of course, we expect the frequency of such events to be much less than this figure for the reasons given above.

We will make a further assumption that all degrees of blockage are equally likely. (In reality, it is expected that modest blockages would occur more often then severe ones, so this is a conservative assumption.) NEDO-10174, Revision 1[288] concluded that 79% blockage would result in DNB and that 98% blockage would result in fuel melting.

$$F(DNB) \leq (0.19)(8.1 \times 10^{-3}/RY) = 1.5 \times 10^{-3}/RY$$

$$F(Melt) \leq (0.02)(8.1 \times 10^{-3}/RY) = 1.6 \times 10^{-4}/RY$$

Consequence Estimate

Localized DNB failures are not severe events. A special CRAC calculation of a rod drop event (in which 770 fuel pins were assumed to fail) gave a result of 7×10^3 man-rem under our usual assumption of 340 people per square mile. Scaling this figure down to one 62-pin fuel assembly, the result of a DNB event becomes 5.6×10^{-4} man-rem. Using the upper limit frequency given in the previous section and assuming a 40-year plant life, the DNB hazard is, at most, about 3×10^5 man-rem/plant and is probably much less. Consequently, DNB failures will not be considered further.

One assembly melting will produce more of a radioactive release, although (according to the GE study) pressure pulses are not expected to be severe. The effects of such an event can be bounded by scaling a BWR-4 release to one assembly. A BWR-4 release is a complete core-melt with enough containment leakage to prevent containment failure due to overpressure. Since one assembly is not very likely to overpressurize the containment, this is a very conservative estimate. In reality, the release would be limited by the high steam line radiation isolation and also by the offgas treatment system isolation on high outlet radiation. Thus, since a typical large BWR has 764 fuel assemblies and a complete core-melt producing a BWR-4 release results in 6.1×10^5 man-rem, we can set a limit on consequences of:

$$R \leq \frac{610{,}000}{764} = 800 \text{ man-rem}$$

Currently, there are 24 BWRs in operation with an accumulated experience of about 370 RY. Additionally, 23 more BWRs are under construction. Assuming a 40-year plant life, there are 590 RY remaining for operating BWRs and 920 RY for future BWRs. This represents a total remaining reactor life of 1,510 RY. Therefore, the total risk reduction associated with this issue is $(1.6 \times 10^{-4})(800)(1{,}510)$ man-rem or 193 man-rem.

Cost Estimate

Industry Cost: PNL postulated[64] an engineering solution consisting of one-way flapper valves to admit coolant into the lower portion of the fuel assembly. Each BWR would have to pay an incremental cost on each new fuel assembly for the additional holes and flapper valves in the lower tie plate. On the average, a plant replaces about 1/3 of its core during annual refueling outages. Assuming 600 and 750 fuel assemblies per core for backfit and forward-fit BWRs, respectively (based on BWR design information), one obtains the following refueling rates:

Backfit BWR: 133 fuel assemblies/RY
Forward-fit BWR: 167 fuel assemblies/RY

Only the incremental cost due to design changes in the assemblies is credited here. Assuming this amounts to $250/fuel assembly, the operation/maintenance costs (interpreted as refueling costs) are as follows:

Backfit BWR: (133 fuel assemblies/RY) ($250/fuel assembly)
= $33,000/RY

Forward-fit BWR: (167 fuel assemblies/RY) ($250/fuel assembly)
= $42,000/RY

For the 24 BWRs in operation, the backfit costs are ($33,000/RY)(590 RY) or $19.5M. Forward-fit costs are ($42,000/RY)(920 RY) or $38.6M for the 23 BWRs under construction. Thus, the total industry cost is $58.

NRC Cost: NRC costs are estimated to be 4 man-weeks for review of the GE topical report, 12 man-weeks for review of engineering solutions, and 1 man-week/plant for supporting implementation. Therefore, the total NRC cost is estimated to be approximately $120,000.

Value/Impact Assessment

Based on an estimated risk reduction of 193 man-rem and a total cost of $58.2M, the value/impact score is given by:

$$S = \frac{193 \text{ man-rem}}{\$58.2\text{M}}$$

$$\approx 3 \text{ man-rem/\$M}$$

CONCLUSION

Based on the above value/impact score, this issue should be placed in the DROP category.

REFERENCES

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission.

288. NEDO-10174, Revision 1, "Consequences of a Postulated Flow Blockage Incident in a Boiling Water Reactor," General Electric Company, October 1977.

380. NEDO-10174, "Consequences of a Postulated Flow Blockage Incident in a Boiling Water Reactor," General Electric Company, May 1970.

381. Memorandum for F. Schroeder from L. Rubenstein, "Review of General Electric Topical Report NEDO-10174, Revision 1," August 18, 1982.

ISSUE 69: MAKE-UP NOZZLE CRACKING IN B&W PLANTS

DESCRIPTION

Historical Background

Cracks were found in the normal make-up high pressure injection (MU/HPI) nozzles of several B&W plants following an inspection of the 8 B&W plants licensed to operate. These cracks appeared to be directly related to loose or missing thermal sleeves. As a result, a B&W Owners' Group Task Force was established to identify the cause of the failures and recommend modifications to eliminate future failures.[513]

The B&W Task Force has completed a generic investigation of the MU/HPI nozzle component cracking problem and has submitted a report containing the findings of that investigation.[514] The report presents relevant facts and probable failure scenarios, as well as recommended modifications to thermal sleeve designs, make-up system operating conditions, and ISI plans. Failure analysis indicated that the cracks were initiated on the inside diameter and were propagated by thermal fatigue. Recent inspections at Midland have also shown that gaps may be present between the thermal sleeve and safe-end in the contact expansion joint. These findings along with stress analysis and testing have implicated insufficient contact expansion of the thermal sleeves as the most probable root cause of the failures. The NRC staff has evaluated the B&W Owners' Group report and has found the report acceptable.

Safety Significance

On the B&W plants with 145, 177, and 205 fuel assemblies, four HPI/MU nozzles (one per cold leg) are used to: (1) provide a coolant source for emergency core cooling, and (2) supply normal make-up (purification flow) to the primary system. In general, one or two of the nozzles are used for both HPI and MU, while the remaining nozzles are used for HPI alone.

The incorporation of a thermal sleeve into a nozzle assembly is a common practice in the nuclear industry to provide a thermal barrier between the cold HPI/MU fluid and the hot high pressure injection nozzle. This helps prevent thermal shock and fatigue of the nozzle. The purpose of the safe-end is to make the field weld easier (pipe to safe-end) by allowing similar metals to be welded. The dissimilar metal weld between the safe-end and the nozzle can then be made under controlled conditions in the vendor's shop. The use of the safe-end also eliminates the need to do any post-weld heat treating in the field. Failures in these HPI/MU nozzles may preclude the proper functioning of the ECCS and/or the normal fluid makeup to the primary system.

Possible Solutions

As a result of their investigation, B&W made the following recommendations[514] as the solutions to the problem:

(1) Reroll the upstream end of the thermal sleeve, when inspections indicate that a gap exists, or repair and/or replace damaged components

(2) Maintain a continuous MU flow greater than 1.5 gpm

(3) Implement an augmented ISI plan

(4) Perform a detailed stress analysis of a nozzle with a modified thermal sleeve design to justify long term operation.

The staff reviewed and evaluated the recommendations of the Task Force and supported all four recommendations. However, the staff concluded that the implementation of Recommendations (1), (3), and (4) were adequate for resolving the issue.

CONCLUSION

All licensees participating in the B&W Owners' Group Task Force performed the repairs to damaged components outlined in Recommendation (1). The augmented ISI program in Recommendation (3) was voluntarily implemented. The stress analysis of Recommendation (4) will be done by the affected licensees and will require an MPA for follow-up staff verification.[667] Thus, this issue was RESOLVED and no new requirements were established.

REFERENCES

513. SECY-82-186A, "Make-up Nozzle Cracking in Babcock and Wilcox (B&W) Plants," July 23, 1982.

514. B&W Document No. 77-1140611-00, "177 Fuel Assembly Owner's Group Safe End Task Force Report on Generic Investigation of HPI/MU Nozzle Component Cracking," Babcock and Wilcox Company, 1983.

667. Memorandum for T. Speis from H. Denton, "Resolution of Generic Issue 69: Make-up Nozzle Cracking in B&W Plants," September 27, 1984.

ISSUE 79: UNANALYZED REACTOR VESSEL THERMAL STRESS DURING NATURAL CONVECTION COOLDOWN

DESCRIPTION

Historical Background

On March 18, 1983, a letter[530] was written from B&W to OIE in which a concern of potential generic safety significance was described relating to an unanalyzed reactor vessel thermal stress that could occur during natural convection cooldown of PWR reactors. The concern emerged from a preliminary B&W evaluation of the voiding event that had occurred in the upper head of the St. Lucie reactor on June 11, 1980. Based upon several conservative assumptions, B&W tentatively concluded that during natural convection cooling there could develop axial temperature gradients between 150° to 200°F in the vessel flange area which could produce thermal stresses in the flange area or in the studs that might exceed code allowables when added to the stresses already considered (bolt-up loads, pressure loads, etc.). B&W acknowledged the preliminary nature of their analysis and noted that, if their conservatively calculated cooling rate of the stagnant coolant in the vessel head (2°F/hr) were to be in the order of 20°F/hr, then the estimated vessel stresses would not be excessive.

B&W requested assistance from the NRC in obtaining any data or information which might help in the technical evaluation of this problem, but noted that this phenomenon is not likely to be a serious near-term safety concern. However, it was their view that it did represent an unanalyzed situation with a potential for margin reduction over plant life. Preliminary evaluations[531,533] by the NRC staff also led to the conclusion that the problem is not a serious near-term safety safety concern although a Board Notification has been made by the staff.[534]

In an RES memorandum[668] to NRR, it was pointed out that the closure flange regions of reactor vessels were of concern in drafting then recent revisions to Appendices G and H of 10 CFR 50; these regions control the pressure-temperature limits at low pressures for vessels that have very little radiation damage in the beltline region. Critical flaw sizes are of the order of ¼ in. deep. The existence of a possible new source of stress in the flange region may have a bearing on the revisions to 10 CFR 50.

In an analysis[669] of the problem by the B&W Owners Group, it was concluded that the concern of this issue has been resolved.

Safety Significance

The safety significance of this unanalyzed reactor vessel thermal stress is that, when added to the existing stresses, the stresses in the flange area or studs may exceed the allowable stress. Moreover, the cycling of these temperatures gradients over the life of the plant may cause a reduction in the fatigue margin or usage factor of the vessel over the life of the plant. In addition, depending upon the vessel temperature distribution, there is a possibility of vessel fracture under these circumstances. These factors could cause vessel cracking leading to unacceptable vessel failure during the life of the vessel. However, it is assumed in this analysis that sufficient water is available to prevent dry-out of the steam generators. Otherwise, the consequences could be more serious than is presently estimated.

Possible Solutions

If these unanalyzed thermal stresses do cause a reduction of fatigue life or lead to vessel stresses that exceed the allowable stresses for the vessel, the solution is assumed to be a slower cooldown rate than the presently allowable rate of 100°F/hr.

PRIORITY DETERMINATION

Frequency Estimate

In an effort to establish the frequency of the occurrence of the unanalyzed stress, it is noted that in the B&W letter[530] is was stated that "...These non-uniform effects in the reactor coolant may occur once the reactor coolant pumps are secured and the decay heat removal system has been actuated in the normal cooldown mode or during natural circulation." This B&W statement is somewhat of an oversimplification, however, and may be misleading insofar as the effects resulting from normal cooldown and the effects of a natural circulation cooldown may be substantially different with regard to the non-uniform effects in question. This is to say, in the normal cooldown mode, the RCPs continue to operate until the reactor and temperature are reduced to the cut-on levels for the decay heat removal system pumps, which are well below the values at which natural circulation would be initiated, if required. Therefore, with the RCPs on during the early phase of the normal cooldown, it would be expected that significant mixing of the fluid in the reactor head would occur in this period of the cooldown so that the non-uniform effects are likely to be minimized if not precluded. The natural convection mode of cooldown, on the other hand, would tend to have larger non-uniform effects because it starts with the highest fluid and material temperatures and the thermal mixing is assumed to be substantially lower than with the RCPs in operation. The frequency corresponding to this cooldown mode will, therefore, be used in this analysis.

A dominant factor that may affect primary coolant pump availability, leading to the need for natural convection cooling, is the loss of off-site power because the large power demand of these pumps usually exceeds the on-site power supply capability. It is noted, however, that Regulatory Guide 1.93[532] provides guidance which permits continued operation of the plant for up to 24 hours after loss of off-site power followed by 6 hours of hot standby, if this implements the safest operating mode whenever the off-site power sources are less than the limiting conditions of operation. For this time period (30 hrs), it is estimated from the information provided in reference 5 that the probability of not recovering off-site power is approximately 0.03. If off-site power did not return in this time the plant is shut down to cold shutdown and natural convection cooling would follow. However, the licensee need not keep the plant up for 30 hours and can elect to shutdown immediately to the cold shutdown state if it is deemed the safest mode of operation following the loss of offsite power. In this case natural convection cooling of the reactor would begin promptly and it is estimated from references 6 and 8 that the maximum thermal gradients in the reactor vessel would occur in approximately 2 to 3 hours. From the information in reference 5 it is estimated that the probability of not restoring off-site electric power within two hours is approximately 0.35. Inasmuch as this latter case represents the more stringent conditions for this issue, the analysis will be based on immediate shutdown to the cold shutdown condition as a conservative scenario.

The frequency of the loss of off-site power is established as 0.118/RY[526] and the probability of not restoring offsite power within 2 to 8 hours is about 0.35. The frequency of attaining the maximum thermal gradients within the reactor as a result of natural convection cooling is estimated to be (0.118)(0.35)/year or 0.04/year.

Consequence Estimate

Following the loss of the primary pumps the reactor will be scrammed. It is assumed here that the on-site AC power is available (otherwise the issue becomes station blackout, which is being treated in USI A-44).[526] Natural convection cooling commences and the core and reactor vessel begin to cool down. Based on the cooling rate of approximately 100°F/hr, it is expected that the large temperature gradients described by B&W[530] will begin to develop after approximately 2 to 3 hours because the vessel head cools down very slowly (~2°F/hr). Inasmuch as the thermal stress is likely to be greatest in the region from just below the vessel flange[530] to the "knuckle" just above the vessel head flange,[535] a crack is postulated to occur in the upper part of the vessel or vessel head if the allowable stress is exceeded following the addition of the thermal stress. As a result, the postulated crack is located above the core region.

Upon development of the crack in the upper part of the reactor vessel, the reactor system will blow down as in a LOCA and the core is likely to become partially or totally uncovered. At this time, 2 to 3 hours after reactor shutdown when the large thermal gradients develop, the decay heat is in the order of 1½% of full power. The time required for the core to slump under these conditions is estimated to be slightly over 1 hour for a 3500 MWt reactor with a core of approximately 1100 ft^3 and a total core heat capacity of 54 BTU/ft^3/°F. Within this time period it is expected that the operator can properly assess the situation and will take appropriate action to prevent core slumping by releasing the water in the accumulators and submerging the core. The integrity of the reactor vessel is expected to be maintained to a level above the height of the core because the high stress regions were limited to the vicinity of the vessel flanges. Boil-off will continue but it is assumed that water replacement from the reactor plant systems will be available to continue to cool the core. Some gap activity release is possible during the period of postulated core uncovery.

This scenario is comparable to PWR Release Category 8 or 9.[16] In these release categories, the core does not melt and only some of the activity in the gaps of the fuel rods is released. However, in PWR Release Category 8 it is postulated that the containment fails to isolate properly and the public dose is estimated to be 75,000 man-rem. In Release Category 9, the containment isolates properly and the public dose is only 120 man-rem which can be neglected. Inasmuch as it cannot be assured that the containment will isolate properly for this issue, despite the fact that the accident is estimated to occur 2 to 3 hours after plant shutdown, it will be assumed that Release Category 8 represents the scenario of this problem. A public dose of 75,000 man-rem is therefore postulated.

From NUREG/CR-2800,[64] the probability that the containment fails to isolate is 0.0073. The potential risk reduction is, therefore, (0.04)(0.0073)(75,000) man-rem/RY or 22.6 man-rem/RY. For an average plant lifetime of 28 years, the total risk reduction is (22.6)(28) man-rem/reactor or 633 man-rem/reactor.

Cost Estimate

The resolution of this problem requires that natural convection cooling proceed at a slower rate than the present allowable rate of about 100° F/hr. It is assumed here that thermal stress condition will be ameliorated if the present cooldown time is increased from approximately 80 hrs. to 160 hrs. Moreover, it is assumed that sufficient water is available to prevent dry out of the steam generators for these time periods. The increase in cooling time represents down-time for the reactor site of an additional 80 hrs or 3 1/3 days. At a cost of $300,000 per day, the additional cost to the plant for each natural convection cooldown is approximately $1M. Over the remaining average life of the plant, it is estimated that 2 to 3 natural convection cooldowns will be required based on the expected frequency of 0.118/RY for the loss of off-site power. The licensee cost associated with three natural convection cooldown events is ($1M)(3)/plant or $3M/plant.

Additional NRC and licensee costs associated with technical studies to be performed for the determination of required cooldown rates, possible technical specification changes, etc., are estimated to be approximately $100,000 per plant. The total cost for resolution of this issue is $3.1M/reactor.

Value/Impact Assessment

Based on total risk reduction of 633 man-rem/reactor and a total implementation and development cost of $3.1M/plant, the value/impact score is given by:

$$S = \frac{633 \text{ man-rem/reactor}}{\$3.1\text{M/reactor}}$$

$$= 204 \text{ man-rem/\$M}$$

CONCLUSION

Based on the value/impact score of 204 man-rem/$M and the potential risk reduction of 633 man-rem/reactor, the priority ranking of this issue is MEDIUM. It is to be noted, however, that these results are based on the assumption that sufficient cooling water is available to prevent dry-out of the steam generators without off-site power. Without this assumption, the consequences would be considerably more serious than estimated above.

REFERENCES

16. WASH-1400 (NUREG-75/014), "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, October 1975.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

526. NUREG/CR-3226, "Station Blackout Accident Analyses (Part of NRC Task Action Plan A-44)," U.S. Nuclear Regulatory Commission, May 1983.

530. Letter to R. DeYoung (NRC) from J. Taylor (B&W), "Unanalyzed Reactor Vessel Thermal Stress During Cooldown," March 18, 1983.

531. Memorandum for R. Vollmer from W. Minners, "B&W Notification Concerning an Unanalyzed Reactor Vessel Thermal Stress During Cooldown," April 7, 1983.

532. Regulatory Guide 1.93, "Availability of Electric Power Sources," U.S. Nuclear Regulatory Commission, December 1974.

533. Memorandum for W. Minners from R. Bosnak, "B&W Notification Concerning an Unanalyzed Reactor Vessel Thermal Stress During Cooldown," April 26, 1983.

534. Memorandum for Chairman Palladino, et al., from D. Eisenhut, "Unanalyzed Reactor Vessel Thermal Stress During Cooldown (Board Notification # BN-83-42)," April 12, 1983.

535. CE-NPSD-154, "Natural Circulation Cooldown, Task 430 Final Report," Combustion Engineering, Inc., October 1981.

536. 86-1140819-00, "Reactor Vessel Head Cooldown During Natural Circulation Cooldown Transients," Babcock & Wilcox Company, February 8, 1983.

668. Memorandum for H. Denton from R. Minogue, "Comments on Generic Issue 79, 'Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown,'" October 5, 1983.

669. Letter to P. Kadambi (NRC) from F. Miller (B&W Owners Group Analysis Committee), "Transmittal of RV Head Stress Evaluation Program Results," October 15, 1984.

ISSUE 81: IMPACT OF LOCKED DOORS AND BARRIERS ON PLANT AND PERSONNEL SAFETY

DESCRIPTION

Historical Background

In October 1982, the EDO appointed a Safety/Safeguards Committee to review NRC security requirements at nuclear power plants with a view toward evaluating the impact of these requirements on operational safety. The Committee issued its findings and recommendations in a report[621] to NMSS on February 28, 1983. The Committee did identify a potential for the security measures at a site to adversely affect safety. In view of this finding, a DL memorandum[542] was issued on May 31, 1983 suggesting that a multi-disciplinary group perform an integrated assessment of this problem with DHFS management in the lead coordinating role. The DL memorandum also suggested that DST prioritize this issue.

Based on the responses to their memorandum, DL advised[623] the NRR Director that the concensus of the responses supported the creation of the multi-disciplinary group. The group's function was to gather the necessary information on this issue and then to prepare a scope of the issue for appropriate consideration. This approach was confirmed[624] by the NRR Director. Subsequently, a DL memorandum[625] to DHFS formally initiated action on this matter.

The multi-disciplinary group began with its first meeting on February 28, 1984 and issued a report on June 8, 1984.[626] Inasmuch as a proposed rule, SECY-83-311,[627] specifically designed to address the security barrier issue had been prepared independently and IE Information Notice No. 83-36[628] also had been issued, the work of the group was limited to non-security barriers.

Safety Significance

The possible failure of locked doors and barriers that may be required for fire protection, radiation protection, flood protection, and administrative controls are of special concern during abnormal or accident situations when emergency conditions may require prompt and unlimited access of the plant operators to safety equipment in order to assure proper plant operation. However, the task group concluded that the locks and barriers associated with these areas could easily be defeated or bypassed in an emergency situation, if necessary, provided there were enough time to take the necessary steps.

Possible Solutions

An evaluation of each plant's locked doors and barriers might be required and appropriate procedural and hardware changes may have to be made in order to establish that operator access is unimpeded during emergency, abnormal, or accident conditions and that prompt operator action is possible as required.

PRIORITY DETERMINATION

Frequency Estimate

The task group considered the likelihood of mechanical failure of the locks used for doors and barriers in nuclear plants to be in the order of 10^{-6}/demand.[629] However, a very conservative estimate of 10^{-4}/demand was adopted. Other failure modes based on personnel error were estimated to be in the order of 10^{-3}/demand, but these kinds of failures were judged to be recoverable within about 15 minutes or less (e.g., lost key or wrong key). Failures of locked doors or barriers that led to delays of 15 minutes or less were considered to be of little safety consequence inasmuch as irretrievable sequences leading to core uncovery would require considerably more time than 15 minutes. In any case, in the event of a failed lock, steps could be taken immediately to defeat the locks used for these general purposes by phyically destroying them with available tools, if necessary, such as drills, crowbars, hammers, etc. Even in the event that a barrier had to be circumvented by destruction of a surrounding reinforced concrete wall, it was estimated that this would take no more than 20 minutes with the proper tools (such as a jack-hammer) which are usually available at plant sites.[626]

Therefore, for the purposes of prioritization, we assumed an initiating event which would require the operator to leave the control room for the purpose of manually operating safety equipment such as loss of DC power supply system with a failure probability of 10^{-2}. Also, we assume a very conservative lock or barrier failure rate of 10^{-4}/demand as well as a failure probability of the operators to bypass the barrier by remedial means within one hour of 0.01. The frequency of potential core-melt under these circumstances is estimated to be approximately $(10^{-2})(10^{-4})(10^{-2})$/RY or 10^{-8}/RY.

Consequence Estimate

Under the above circumstances, even with the maximum public dose estimated at about 5×10^{6} man-rem (see Introduction), the risk would be calculated to be in the order of $(10^{-8}/\text{RY})(5 \times 10^{6}$ man-rem)(28 years)=1.4 man-rem/reactor, or less, assuming an average remaining reactor lifetime of 28 years. It is to be noted that, even if the failure rate of the operators to recover were to be increased by a factor of ten, the large element of conservatism in the mechanical failure rate of the locks would tend to offset this increase. Moreover, it is likely that the public dose would tend to be somewhat lower as well, so that the estimated risk is likely to represent an upper bound of this issue.

Cost Estimate

Based on the deliberations of the task group, the estimated cost to evaluate and make modifications to each plant and its procedures is approximately \$1.7M/plant.[626] This cost is based on the following factors:

(1) A one-time evaluation of existing plant locked doors......\$ 200,000
and barriers

(2) Resolution of adverse safety findings....................$1,000,000
[Cost for maintaining keys for a security force of 24 per plant is estimated to be $21,000/reactor.[627] To provide additional keys for operators (5 per shift for 5 shifts) for a reactor lifetime of 28 years is approximately $612,500. Cross-training of security and operational personnel based on 50 operators and security personnel for 1 day/year/plant, over the lifetime of the plant (28 years) is assumed to be (1/365)(50)(28)($100,000) = $391,232].

(3) Ongoing program to ensure future reduction................$ 280,000
of safeguards impact on safety ($10,000/year for an average reactor lifetime of 28 years)

(4) NRC reviews of plant modifications........................$ 200,000

TOTAL.................$1,680,000

Value/Impact Assessment

Based on the estimated risk reduction of 1.4 man-rem/reactor and the estimated cost of $1.7M/plant to effect this reduction in risk, the value/impact score is given by:

$$S = \frac{1.4 \text{ man-rem/reactor}}{\$1.7\text{M/reactor}}$$

$$= 0.82 \text{ man-rem/\$M}$$

Other Considerations

It is noted that the priority ranking of this issue is relatively insensitive to reasonable changes in the factors comprising the risk estimate. On the other hand, a consideration that has not been accounted for in this evaluation is the adverse effect of locked doors and barriers on sick or injured plant personnel. Locked doors or barriers may delay the administering of emergency aid to injured personnel in a timely way. This consideration would tend to raise the priority ranking, but the combined frequency of illness or injury and a significant delay from a failed barrier seems to be too low to justify a significant increase in the priority ranking.

CONCLUSION

Based on the calculated value/impact score of 0.82 man-rem/$M, the estimated risk reduction of 1.4 man-rem/reactor, and the other considerations noted above, it is concluded that this issue warrants a priority ranking of DROP.

REFERENCES

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623. Memorandum for H. Denton from D. Eisenhut, "Potential Safety Problems Associated with Locked Doors and Barriers in Nuclear Power Plants," December 22, 1983.

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627. SECY-83-311, "Proposed Insider Safeguards Rules," July 29, 1983.

628. IE Information Notice No. 83-36, "Impact of Security Practices on Safe Operations," U.S. Nuclear Regulatory Commission, June 9, 1983.

629. Memorandum for the Record from L. Bush, "Probability of Failure of Locks," May 24, 1984.

ISSUE 86: LONG RANGE PLAN FOR DEALING WITH STRESS CORROSION CRACKING IN BWR PIPING

DESCRIPTION

Historical Background

In March 1982, leaks were detected in the heat-affected zones of the safe-end-to-pipe welds in two of the 28 in. diameter recirculation loop safe ends at Nine Mile Point Unit 1. Subsequent UT revealed extensive cracking at many weld joints in the recirculation system. The cause of the cracking was determined to be intergranular stress corrosion cracking (IGSCC).

Although cracking in large diameter piping had been found previously in Germany and Japan, the finding at Nine Mile Point Unit 1 was the first known U.S. occurrence of IGSCC in large piping (pipe diameters > 10 in.). IE Information Notice 82-39[608] was issued on September 21, 1982 to alert all BWR licensees to the problem. The staff held meetings with GE, EPRI, and BWR Owners to discuss the relevance of the Nine Mile Point cracking to other BWRs. The near-term inspection of welds in large-diameter recirculation piping was discussed at a meeting of the staff with BWR licensees on September 27, 1982. Following this, IE Bulletin 82-03[609] was issued on October 14, 1982 and required the 8 BWRs that were scheduled for outages through January 31, 1983 to perform inspections of a reasonable sample of the recirculation system welds during their respective outages.

After cracking in large-diameter piping was observed in 5 of the first 7 plants inspected in response to IE Bulletin 82-03,[609] the staff issued IE Bulletin 83-02[610] to extend the inspection requirements to all other BWRs. On August 1, 1983, the EDO established a Piping Review Committee to investigate specific incidents of pipe cracking at all plants with emphasis to be placed on IGSCC that had been reported in the recirculating systems of BWRs. Under the auspices of the NRC Piping Review Committee, a Task Group on Pipe Cracking was convened to develop a long-range plan for dealing with IGSCC.

The status and results of inspections of piping welds for IGSCC conducted at various operating BWRs were reported to the Commission in SECY-83-267,[612] SECY-83-267A,[613] SECY-83-267B,[614] SECY-83-267C,[615] SECY-84-9,[616] SECY-84-9A,[617] and SECY-84-166.[618] The short-term reinspection and repair criteria were issued to BWR licensees in Generic Letter 84-11[620] on April 19, 1984, and were to be used in inspections subsequent to the issuance of IE Bulletins 82-03[609] and 83-02.[610] The Task Group report (NUREG-1061,[611] Volume 1) was drafted in April 1984 and submitted to the Commission in July 1984 with SECY-84-301,[619] the staff's long-range plan for dealing with IGSCC in BWR piping. NUREG-1061,[611] Volume 1, which includes value/impact analyses for the four possible solutions discussed below, was published in August 1984.

Safety Significance

Pipe cracking resulting from IGSCC can cause a LOCA which, in turn, contributes to core-melt frequency.

Possible Solutions

The results of the Task Group study indicate that there are four possible solutions for preventing IGSCC in BWRs:

(1) Piping replacement without hydrogen water chemistry (HWC) - considered to be a long-term fix.
(2) Induction heating stress improvement and HWC - considered to be a long-term fix.
(3) Augmented inspection, weld repair, and HWC - considered to be a partial intermediate fix.
(4) Augmented inspection and weld repair without HWC - considered to be only a short-term fix.

CONCLUSION

The staff's long range plan for resolving the problem calls for the following actions:

(1) Issuance of NUREG-1061,[611] Volume 1.
(2) Incorporation of the Task Group recommendations that are implementable at this time into NUREG-0313, Revision 2, and issuance of this document.
(3) Preparation of a generic letter that incorporates NUREG-0313, Revision 2, and issuance of this letter to all BWR licensees requesting their proposals for bringing their plants into compliance with 10 CFR 50.55a(g).
(4) Pursuance with the appropriate industry Code Committees changes in the areas of NDE personnel qualification and inspection procedures, to bring ASME XI[14] requirements into conformance with the staff's recommendations.

Thus, the solution to this issue is available.

REFERENCES

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609. IE Bulletin 82-03, "Stress Corrosion Cracking in Thick-Wall Large Diameter, Stainless Steel Recirculation System Piping at BWR Plants," U.S. Nuclear Regulatory Commission, October 14, 1982.

610. IE Bulletin 83-02, "Stress Corrosion Cracking in Large Diameter Stainless Steel Recirculation System Piping at BWR Plants," U.S. Nuclear Regulatory Commission, March 4, 1983.

611. NUREG-1061, "Report of the U.S. NRC Piping Review Committee - Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Boiling Water Reactor Plants," U.S. Nuclear Regulatory Commission, August 1984.

612. SECY-83-267, "Status Report on Observation of Pipe Cracking at BWRs," July 1, 1983.

613. SECY-83-267A, "Update of Status Report on Observation of Pipe Cracking at BWRs (SECY-83-267)," July 11, 1983.

614. SECY-83-267B, "Update of Status Report on Observation of Pipe Cracking at BWRs (SECY-83-267 and 267A)," August 8, 1983.

615. SECY-83-267C, "Staff Requirements for Reinspection of BWR Piping and Repair of Cracked Piping," November 7, 1983.

616. SECY-84-9, "Report on the Long Term Approach for Dealing with Stress Corrosion Cracking in BWR Piping," January 10, 1984.

617. SECY-84-9A, "Update of Status Report on BWR Pipe Cracks and Projection of Upcoming Licensee Actions," January 27, 1984.

618. SECY-84-166, "Update of Status Report on BWR Pipe Cracks and Projection of Upcoming Licensee Actions," April 20, 1984.

619. SECY-84-301, "Staff Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping," July 30, 1984.

620. NRC Letter to All Licensees of Operating Reactors, Applicants for Operating License, and Holders of Construction Permits for Boiling Water Reactors, "Inspections of BWR Stainless Steel Piping," (Generic Letter 84-11), April 19, 1984.

ISSUE 90: TECHNICAL SPECIFICATIONS FOR ANTICIPATORY TRIPS

DESCRIPTION

Historical Background

Reactor protection systems (RPS) or "scram" systems are tripped by many diverse signals. The purposes of these various signals can be broadly divided into three classes: protection of the reactor core (e.g., overpower signals), protection of major components (e.g., vessel overpressurization signals) and anticipatory trips. The purpose of an anticipatory trip signal is to scram the reactor at the very beginning of a transient, and thus minimize the degree of upset of the plant and avoid actuation of engineered safety features.

By definition, an anticipatory trip is not taken credit for in the plant's safety analyses, even in the sense of satisfying a single failure criterion. (Conversely, if a transient analysis calculation, upon which technical specifications are based, takes credit for early reactor scram due to an "anticipatory" trip, the trip can no longer be considered "anticipatory.") Originally, it was AEC (and then NRC) regulatory policy to consider such trips to be installed for the licensees' convenience rather than for licensing purposes.[630] Thus, no technical specification requirements were placed upon these trips.

At the time the STS was introduced, this policy was changed and all trips in the RPS were included in the STS. However, plants licensed prior to the advent of STS were not backfitted with technical specifications on anticipatory trips. In addition, post-STS reviews of custom technical specifications done after the STS were introduced did not require the inclusion of specifications on anticipatory trips.

This particular generic issue originated when Region II noted the anomaly and found that licensees were often performing little or no maintenance on these trips in the absence of technical specification requirements.[630] With no mention of these trips in the technical specifications, Region II could take little action even in cases where inspectors had reason to doubt the operability of these trips.

DSI commented that, since there is a broader generic issue involving the overall adequacy of the technical specifications, this issue should be subsumed in the broader issue. RSB is planning a systematic study of all technical specifications, but has not yet defined a scope or schedule for such a study. If the broader study is begun, this issue can be considered as part of it, but no purpose would be served by delaying prioritization of this issue.

Safety Significance

The suspected safety deficiency identified[630] was that, "because anticipatory trips are a part of the protection system, a failure or maintenance action in the anticipatory trip could cause other trips relied on in the accident analysis to be degraded below an acceptable level." The design of the RPS, however, leans strongly towards a fail-safe direction, i.e., failure of any channel may cause an inadvertent reactor scram, but should never prevent a trip of another channel from scramming the reactor. Questions of this nature are valid (e.g., can the W scram-on-turbine-trip interact with other channels via the P-7 interlock?), but the DL memorandum[630] does not provide any specific concerns. Moreover, ICSB mentioned none in its examination of this issue,[631] and the fault tree analysis of WASH-1400[16] (Appendix II § 5.2) found no such interaction. In the absence of specific concerns, this part of the issue is better treated under the auspices of the ATWS program. It will not be treated further here.

The second area of potential safety significance lies in a plant's response to a transient. Since anticipatory trips reduce the degree of plant upset, they also (in principle) reduce the frequency of challenges to engineered safety features, and thus (again, in principle) reduce the frequency of transient-initiated accidents.

Possible Solutions

The proposed solution for this item is implicit in its definition - impose technical specification requirements on anticipatory trips. Such specifications would include: (1) limiting safety system settings (LSSS), which provide setpoints; (2) limiting conditions for operation (LCO), which require the equipment to be operable during appropriate operational modes; and (3) surveillance requirements (SR).

PRIORITY DETERMINATION

A search through some older technical specifications uncovered the following anticipatory trips:

(1) PWR - High neutron flux, source range
PWR - High neutron flux, intermediate range
(2) PWR - Turbine trip
(3) PWR - Low steam generator level coincident with steam-feed mismatch
(4) BWR - High neutron flux, source range

Because these trips are notable via their absence in the technical specifications, there is no guarantee that this list is complete. Nevertheless, these anticipatory trips should be representative. There are other trips (e.g., PWR RCP breaker open) which have LCOs and SRs, but do not appear in the LSSS. Historically, this is because it is difficult to define "setpoints" for such trips. This is not a safety but a licensing improvement (enforceability) aspect and will not be considered here.

Normally, calculations of scram failure probabilities are done for each channel of the RPS, explicitly accounting for the various 1/2, 2/3 and 2/4 logic matrices with estimates of common mode failure rates included. Here, we can make some simplifying assumptions.

First, imposing technical specifications on anticipatory trips does not affect the common mode failure rate of the RPS. Thus, when the changes in failure probabilities are calculated, the common mode failures are subtracted out.

Second, lack of maintenance of an anticipatory trip is a common mode of both trip channels associated with the anticipatory trip parameter, i.e., if one channel fails, the other is also quite likely to fail. In the calculations, we will assume that the probability that a trip signal will fail to cause a scram (i.e., both channels failing) is 0.01/demand on an anticipatory trip which is not in the technical specifications, based on judgment but augmented by conversations with the senior inspector who originated this issue. All other trip parameters will be assumed to have failure rates of 1×10^{-4}/demand, based on Appendix II § 5.2 of WASH-1400.[16]

Frequency/Consequence Estimate

(1) PWR - High Neutron Flux, Source and Intermediate Range

These trips are functional only during reactor startup. They are backed up by the low setpoint of the power range neutron flux channels. Usually, the source range monitor setpoint is set at 10^5 counts/second (which is equivalent to roughly 0.01% of rated reactor power) and the intermediate range setpoint and power range low setpoint are both set at 25% of rated power. Thus, the source range is truly anticipatory in the sense of attempting to stop a transient early while the intermediate range backs up, but does not anticipate, the low power setpoint of the power range channels.

Rod Ejection: The first "transient" of interest is not an anticipated operational occurrence, but is considered an accident. A rod ejection is a reactivity excursion which is, if significant at all, too rapid for the anticipatory nature of the source range trip to make much difference. Thus, the safety contribution comes from increased scram reliability, not early scram. We will make the following assumptions:

10 reactor startups/RY[178]
2 days of vulnerability/startup (based on judgment)
1×10^{-5} rod ejections/RY (based on WASH-1400,[16] App. I § 4.3)

This works out to 5.5×10^{-7} rod ejections/RY at low power. The change in scram failure rate is $[(1 \times 10^{-2}) - (1 \times 10^{-4})]$ for the source and intermediate range trips, multiplied by (1×10^{-4}) for the low setpoint power range trip. The consequences are those of a partial core-melt (if all these trips fail) which we will bound with those of a PWR-5 release (core-melt with the containment not isolated). Using the assumption of a uniform population density of 340 persons/square mile, a 50-mile radius, a central mid-west plain meteorology and no ingestion pathways, a PWR-5 release results in 1×10^{-6} man-rem. The result is:

$$\Delta F = (5.5 \times 10^{-7} \text{ startup ejections/RY}) [(1 \times 10^{-2})^2 - (1 \times 10^{-4})^2] (1 \times 10^{-4})$$
$$= 5.5 \times 10^{-15} \text{ partial core-melt/RY}$$

$$\Delta FR \leq (5.5 \times 10^{-15} \text{ releases/RY}) (1 \times 10^6 \text{ man-rem/PWR-5 release})$$
$$\leq 5.5 \times 10^{-9} \text{ man-rem/RY.}$$

Short Periods: Under certain conditions of core burnup and high xenon inventory, differential rod worth tends to concentrate in a relatively narrow vertical range in the core during startup. This effect is much more pronounced in a BWR core (see Issue 6), but can also occur in PWRs.

Should this occur, the reactor core will go suddenly from subcritical to supercritical with a rapid positive period (on the order of 10 seconds). Older plants do not have flux rate trips. If the SRM trip fails and the operator does not terminate the transient manually (assume 10% chance), the reactor core will not be shut down until power reaches the 25% intermediate and power range setpoints. Fuel failure could occur due to pellet/cladding interaction (PCI) during the rapid power ascension, or due to DNB because of the highly axially peaked power shape.

To the best of our knowledge, no such events have occurred at a PWR. IE Bulletin No. 79-12[6] and IE Circular No. 77-07[5] list 5 such events in BWRs as of May 31, 1979. This corresponds to about 155 BWR-years of experience. We will assume that the PWR frequency is at most one-tenth of the BWR frequency, or 3×10^{-3} event/PWR-year. We will also bound the consequences with those of PWR-8 and PWR-9 releases, which correspond to mitigated large break LOCAs without and with containment isolation (i.e., widespread cladding failure but no fuel melting). The WASH-1400[16] assumption of containment isolation failure probability is 0.1. The risk associated with the short period scenario then is:

$$\begin{aligned}\Delta FR \leqq\ & (3 \times 10^{-3}\ \text{event/RY})\ [1 \times 10^{-2}) - (1 \times 10^{-4})\ \text{SRM trip failures/event}] \\ & \times (0.1\ \text{operator failures/event}) \\ & \times [\ (120\ \text{man-rem/accident, containment isolated}) \\ & \quad + (0.1\ \text{isolation failures/accident})\ (7.5 \times 10^{4}\ \text{man-rem/accident,} \\ & \quad \text{containment not isolated})] \\ \leqq\ & 2.3 \times 10^{-2}\ \text{man-rem/RY.}\end{aligned}$$

<u>Rod Bank Withdrawal Error</u>: This transient is characterized by a slower reactivity insertion rate than those of the transients discussed above. Thus, fuel failure is not likely to occur because of a high rate of power ascension at the beginning of the transient, but instead may occur due to DNB as the core comes into the power range, possibly with an adverse power distribution due to some rod banks remaining in the core.

For this to happen, the source, intermediate, and power range trips must fail. In addition, the rod stop must fail. We will assume a failure rate of 0.1 (rather than 0.01) for the rod stop, since it is associated with the intermediate range detectors. We will not assume credit for operator action, since it is probably the operator who is causing the event. Again, as discussed in the short-period transient, we will bound the consequences with those of PWR-8 and PWR-9 releases, assuming a containment isolation failure probability of 0.1.

WASH-1400[16] estimates the frequency of PWR uncontrolled rod withdrawal transients to be 0.01/RY. We will assume that half of these occur during startup maneuvers. The risk estimate is then:

$\Delta FR \leqq$ (0.01 rod withdrawal events/RY)
x (0.5 percentage in startup)
x (1×10^{-2} - 1×10^{-4} source range failure rate change)
x (0.1 rod stop failure rate)
x (1×10^{-2} - 1×10^{-4} intermediate range failure rate change)
x (1×10^{-4} low setpoint power range failure rate)
x [(120 man-rem/PWR-9 release) + (0.1 containment failure rate) (7.5×10^{4} man-rem/PWR-8 release)]

$\leqq 4 \times 10^{-8}$ man-rem/RY.

For this transient to progress to core-melt, the high pressurizer pressure, high pressurizer water level, overtemperature and overpower ΔT, and several other trips must fail. However, the DNB event frequency above is already down to about 5×10^{-12}. Thus, even if there were no credit for these additional trips and the core-melt resulted in the worst case consequences (5.4×10^{6} man-rem from a PWR-1 release, where the core-melt causes a steam explosion which ruptures both the reactor vessel and the containment), the resulting public risk would be only 3×10^{-5} man-rem/RY.

Boron Dilution: This transient is caused by a CVCS malfunction which dilutes soluble boron in the reactor moderator. The reactivity insertion rate is very slow, on the order of 10^{-5}/second at the start and diminishing asymptotically to zero as the moderator becomes more dilute. Thus, the transient is also very slow, giving the operator as much as an hour or more to take action for a dilution event during startup. Because the power increase is slow, fuel failures due to PCI are not expected.

For a PWR core at BOC conditions, there is insufficient reactivity worth in the control rods to maintain the core subcritical with no soluble boron in the moderator. Thus, a reactor scram does not permanently terminate the transient; operator action is necessary to stop the dilution and re-borate the moderator. A reactor scram does give the operator more time to respond, however.

EPRI NP-801[178] estimates a frequency of 0.03 event/RY for boron dilution. Again, we will assume that almost half of these events occur during startup. As the event progresses, the operator (for whom we will allow credit for this slow transient) must fail to observe the transient and take no action (assume probability $\leqq$ 0.10). The source range trip must fail (0.01 - 0.0001), the intermediate range trip must fail (0.01 - 0.0001) and the low setpoint power range trip must fail (0.0001). At this point, the estimated change in frequency is down to 1.5×10^{-11}/RY. Reactor power increases past 25% and thermal energy is dumped via the main condenser steam dump, the ADVs and/or the steam generator safety valves, depending on plant conditions. Eventually, as reactor power increases to a value too great to be dissipated by these means, trip signals on pressurizer high pressure, pressurizer high water level, steam generator low-low water level, turbine trip (which functions as a 50% power trip as the P-9 permissive is reached), overtemperature ΔT, and several others may scram the reactor. (Fuel still has not been damaged.) However, even with no credit for these non-neutron-flux trips, a frequency of 1.5×10^{-11}/RY is estimated. Thus, this transient will not be considered further.

(2) PWR - Turbine Trip

A turbine trip, normally sensed as either two out of three low autostop oil pressure signals or four out of four turbine stop valves closed, causes a reactor scram when the plant is operating above a preset power level (e.g., 10%). If a turbine trip occurs and this scram fails, steam pressure will rise in the secondary system. The atmospheric dump valves and steam generator safety valves will be available to limit the pressure rise, and the steam dump (which is usually in T_{avg} mode during power operation) will open and dump steam directly to the condenser. However, before these alternate energy sinks become available, the primary system will experience a rapid heat up. Expansion of the primary coolant will force coolant into the pressurizer, compressing the steam bubble. Reactor scram signals will be generated by the high pressurizer pressure, over-temperature ΔT and high pressurizer level signals, and the transient will "turn around." Pressurizer spray will also turn on to limit the primary pressure transient, but it is likely that the PORVs will open.

We will restrict this discussion to W and CE plants. Upgraded anticipatory reactor trips on turbine trips were required of B&W plants by TMI Action Plan Item II.K.2(10). This item, implemented under MPA F-28, is now complete.

The safety significance of this trip is two-fold: (1) the anticipatory trip increases the reliability of the entire reactor protection system, thus decreasing the frequency of ATWS events, and (2) by preventing opening of a pressurizer PORV, the frequency of a small LOCA is also decreased.

EPRI NP-801[178] lists the following transient frequencies, all of which involve a turbine trip:

Turbine Trip	1.48/RY
Load Rejection	0.45/RY
Loss of Condenser Vacuum	0.12/RY
Loss of Circulating Water	0.07/RY
Total:	2.12/RY

The last two initiators will also disable the steam dump and, on plants with turbine-driven main feedwater pumps, cause a loss of feedwater. However, tripping of the main turbine is the first event to cause a reactor transient.

We will assume that a PORV opens about half the time in these transients. Also, WASH-1400[16] estimates the probability that a PORV will fail to re-close to be 1% per actuation (WASH-1400,[16] App. V § 4.3.1). The S2 LOCA sequence is then a turbine trip transient (2.12/RY), failure of the anticipatory trip channels $[(1 \times 10^{-2}) - (1 \times 10^{-4})]$ opening of a PORV (0.5), failure of the PORV to re-close (0.01) and failure of the operator to correctly diagnose the problem and close the PORV block valves (assume 0.1). The change in S2 frequency is then 1×10^{-5} RY.

The scram reliability is increased by the diversity of an anticipatory trip. However, placing technical specifications on the anticipatory trip does not affect the common mode failure rate of the RPS. Let CM denote the common mode failure rate. If the high pressurizer pressure, overtemperature ΔT, and high pressurizer level signals have failure rates of 0.0001/demand, and the imposition of technical specifications reduces the turbine trip signal failure rate from 0.01 to 0.0001, the RPS failure rate is:

$(0.0001)^3$ (0.01) + CM without anticipatory trip TS
$(0.0001)^3$ (0.0001) + CM with anticipatory trip TS.

The change in RPS failure rate is 9.9×10^{-15}; the common mode contribution subtracts out. This is negligible compared with the S2 sequences, therefore it will not be considered further.

To get the public risk associated with the S2 sequences, we simply normalize the WASH-1400[16] results (which assumed an S2 frequency of 1×10^{-3}/RY) to a frequency of 1×10^{-5}/RY (see WASH-1400[16] Table V 3-14):

Release Category	Normalized ΔS2 Freq. (RY^{-1})	Consequence (man-rem)	ΔFR (man-rem/RY)
PWR-1	1.0×10^{-9}	5.4×10^{6}	5.4×10^{-1}
PWR-2	3.0×10^{-9}	4.8×10^{6}	1.4×10^{-2}
PWR-3	3.0×10^{-8}	5.4×10^{6}	1.6×10^{-1}
PWR-4	3.0×10^{-9}	2.7×10^{6}	8.1×10^{-3}
PWR-5	3.0×10^{-9}	1.0×10^{6}	3.0×10^{-3}
PWR-6	2.0×10^{-8}	1.5×10^{5}	3.0×10^{-3}
PWR-7	2.0×10^{-7}	2.3×10^{3}	4.6×10^{-4}
Total:	2.6×10^{-7}		2.0×10^{-1}

(3) PWR - Low Steam Generator Level Coincident with Steam-Feed Mismatch

This anticipatory trip will scram the reactor on low steam generator water level coincident with steam flow greater than feedwater flow by a preset amount (usually 40% of rated). It is backed up by the low-low steam generator water level trip.

The initiating event here is a total loss of feedwater event in any steam generator. Partial loss of feedwater events or total loss of feedwater flow at reduced power levels may not produce sufficient mismatch between steam and feedwater flow to actuate the anticipatory trip. Of course, such events are also slower and early scram is not as important.

If all feedwater pumps are lost, the secondary side water temperature will rise because of the loss of the relatively cool feedwater, the heat transfer across the steam generator tubes is reduced and the primary side heats up. Simultaneously, the secondary water level decreases. The anticipatory trip signal will occur, and then the low-low steam generator level trip. The increasing primary temperature and resultant coolant swell will force more coolant into the pressurizer, compressing the steam bubble and causing an increase in primary system pressure. Reactor trip signals on overtemperature ΔT, high pressurizer pressure and high pressurizer level will occur if the reactor has not already been scrammed, and the PORVs may open to limit the pressurizer pressure. The AFW system will also be initiated by the loss of main feedwater pumps or by low-low steam generator water level. If the AFW system fails and the steam generator tubes uncover, primary side temperature and pressure will rise more rapidly and the pressurizer safety valves will open. However, the probability of this event is not greatly affected by reliability of the anticipatory scram. Thus, it will not be considered here.

As in the turbine trip transient evaluated previously, the anticipatory trip decreases the probability of an ATWS event and also helps prevent a pressurizer PORV from opening, thus decreasing the frequency of a small LOCA.

EPRI NP-801[178] lists several transients which will cause a loss of feedwater:

Loss of feedwater (one loop)	0.99/RY
Loss of feedwater (all loops)	0.08/RY
Feedwater instability (operator error)	0.66/RY
Feedwater instability (mechanical problem)	0.50/RY
Loss of one condensate pump	0.05/RY
Loss of all condensate pumps	0.00/RY
Total:	2.28/RY

As in the turbine trip transient, we will assume that a PORV opens half the time if the anticipatory trip fails and that the PORV fails to re-close 1% of the time. The S2 frequency is then 2.28 transients/RY multiplied by the change in probability of anticipatory trip failure (0.01 - 0.0001), the probability of PORV opening (0.5), the probability of failure of the PORV to re-close (0.01), and the probability of operator failure to close the block valve (0.1). The result is 1×10^{-5}. Fortuitously, this is the same as the turbine trip case and thus the risk figures are the same:

2.6×10^{-7} core-melt/RY   0.2 man-rem/RY

The change in the probability of RPS failure is given by the change in anticipatory trip failure (0.0099) multiplied by the non-common mode failure probabilities of the trips on low-low steam generator level, over-temperature ΔT, high pressurizer pressure and high pressurizer level, each of which is 0.0001. The resulting change in ATWS frequency is on the order of 2.3×10^{-18}/RY which is negligible compared to the S2 LOCA considerations above.

(4) BWR - High Neutron Flux, Source Range.

As in the PWR case, BWR licensing basis transient calculations for startup events do not take credit for the source range monitor (SRM) and intermediate range monitor (IRM) trip setpoints, but instead assume that the reactor is scrammed by the startup mode setpoint of the average power range monitor (APRM) system, usually 15% of rated power. Unlike the PWRs, the IRM scram setpoints are already in the technical specifications; the SRM scram setpoints are not required (although the monitoring function of the SRM is addressed). Moreover, it is common (if not universal) practice to disable the SRM scram inputs with shorting links after the plant's initial core loading is complete. Thus, the SRM scram generally has a failure probability of 1.

However, the SRM scram, at its usual setpoint of 5×10^5 counts/second, is generally not the first scram to occur during a startup transient or accident. The reason is that, in a BWR, the IRM rod block and scram setpoints are defined as percentages of full scale for each IRM range, and the IRM ranges cover five decades. The SRMs and IRMs must overlap. SRMs are interlocked such that they cannot be withdrawn unless the IRMs are on Range 3 or above. If the IRMs are on Range 1 (and if they are not, a rod block on IRM Downscale will prevent rod withdrawal), the IRM scram will occur virtually simultaneously with or (more likely) prior to the SRM scram.

Therefore, the SRM scram will not reduce plant upset. It will only somewhat increase the reliability of the RPS. The events of interest are the rod drop accident and the short period transient. (Because of the individual rod pulls used in a BWR, there is no analog to the PWR rod bank withdrawal error.)

Rod drop accidents and probabilities are discussed in an RDA Statistical Analysis.[632] The basic sequence starts with 2×10^3 rod withdrawals/RY in the startup range. To get a rod drop accident, a rod must disconnect (max 2×10^{-4}) become stuck (max 1×10^{-2}), and become unstuck and drop at the appropriate (we should say inappropriate) time (max 6×10^{-3}). (This does not imply that the rod is out of sequence or will lead to high worth.) This works out to 2.4×10^{-5} rod drop accidents/RY requiring a reactor scram. The maximum improvement the SRM scram channel can make is $(1 - 0.0001)(0.0001)^2$, or 1×10^{-8}. The ΔF involved is then 2.4×10^{-13} event/RY. This is a negligible frequency. Even if such an event ruptured both the vessel and containment and completely melted the entire core under the worst conditions (i.e., BWR-2 release), the maximum public risk would be 1.7 micro-man-rem/RY.

Short period events are more common. As was mentioned earlier, IE Bulletin No. 79-12[6] and IE Circular No. 77-07[5] list 5 such events in 155 BWR-years, which is a frequency of 3.2×10^{-2}/RY. These events will happen with the IRM channels set on their first range and the SRM scram, if functional, would not anticipate other scrams but instead would provide a backup to the IRM scram. Rod blocks are largely ineffective here since the high incremental rod worth is tied up in one 6-in. notch.

Historically, these events have occurred just at the point of criticality. Since the operator is using the period meters to detect criticality, a short period event is easily noticed, and these events have generally been terminated by operator action, not by the RPS. If the operator does not intervene (assume probability of 0.10) and the IRM and APRM scrams fail (probability 1×10^{-8}), the reactor core would ascend into the power range, where the usual reactivity coefficients would turn the transient around. The only consequence would be some cladding failure due to PCI. The consequences to the public can be bounded by those of a licensing basis rod drop accident in which 770 fuel rods fail. These consequences are 0.007 man-rem/event. Thus, the net risk from short period events, even with no credit for the SRM scram, is at the most:

$$(3.2 \times 10^{-2}/RY)(0.10)(1 \times 10^{-8})(0.007 \text{ man-rem})$$
$$= 2.2 \times 10^{-13} \text{ man-rem/RY}$$

Again, this is negligible.

Based on the calculations in (1), (2), and (3) above, the core-melt/RY frequency estimate is:

$$(5.5 \times 10^{-15}) + (2.6 \times 10^{-7}) + (2.6 \times 10^{-7}) = 5.2 \times 10^{-7}$$

Consequence Estimate

Based on the calculations in (1), (2), and (3) above, the public risk reduction is estimated to be:

$$[(5.5 \times 10^{-9}) + (2.3 \times 10^{-2}) + (4 \times 10^{-8}) + (2 \times 10^{-1}) + (0.2)] \text{ man-rem/RY}$$

$$= 0.42 \text{ man-rem/RY}$$

Approximately 20 PWRs would be affected by the proposed action. (Action on BWRs is unlikely to be approved since the potential safety gain from the SRM trip is so small.) Assuming an average remaining life of 20 years for the affected plants, the total remaining operating life is 400 RY. Thus, the total public risk reduction associated with this issue is approximately 170 man-rem.

Cost Estimate

Technical specifications on anticipatory trips would probably be similar to those now in the PWR STS. This would involve for Source Range Neutron Flux: (1) channel checks every shift except during power operation, (2) calibration every refueling outage, and (3) analog operational tests monthly and prior to startups. For Intermediate Range Neutron Flux this would include: (1) channel checks every shift while in startup, (2) calibration every refueling outage, and (3) analog operational test monthly and prior to startup. For steam generator low level/steam-feed mismatch this would include: (1) calibration every refueling outage, and (2) analog operational tests monthly. For turbine trip this would include trip actuation device tests prior to startup. We will assume: 10 startups/year, 2 months refueling outage every 18 months, 5 days to go from cold shutdown to power, 10 minutes for channel checks, 4 hours for calibrations, and 1 hour for analog and trip device tests. This results in an expense of about $4,000/RY at a manpower cost of $100,000/man-year.

We will also assume that monitoring compliance will take 8 hours/RY of inspection time. Finally, we will assume that two full man-years of effort will be needed to develop and carry out an action plan for this item, including developing CRGR packages, etc. The total cost for implementing the solution to this issue is $98,000/reactor or approximately $2M for all reactors.

Value/Impact Assessment

Based on a public risk reduction of 170 man-rem and a cost of \$2M, the value/impact score is given by:

$$S = \frac{170 \text{ man-rem}}{\$2M}$$

$$= 85 \text{ man-rem}/\$M$$

Other Considerations

The above figures above do not include credit for averted cleanup costs to the licensee. It should be remembered that avoidance of PORV opening (and secondary side safety valve opening) is a major reason for installing anticipatory trips. Inclusion of averted cleanup costs as a credit against the cost to the licensee could significantly raise the priority score. Moreover, the occupational man-rem averted by preventing PORV opening (and possible rupture of the pressurizer relief tank rupture disc) might also be significant.

CONCLUSION

Based on the low safety significance and low value/impact score, a LOW priority is recommended for this issue.

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631. Memorandum for F. Miraglia from W. Houston, "Task Interface Agreement Task No. 83-77 (TAC 40002, PA-157)," November 29, 1983.

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ISSUE 92: FUEL CRUMBLING DURING LOCA

DESCRIPTION

Historical Background

Experiments conducted at several test facilities during the past few years have shown that irradiated fuel can fragment (crumble) into small pieces during a LOCA. Some evaluation of this effect has been made for NRC by EG&G.[622] Although it was concluded that temperature increases due to relocation of crumbled fuel would be smaller than those due to other processes that are now treated conservatively in Appendix K, the question was raised whether this effect should be treated as a long-term generic safety issue.[622]

Safety Significance

During the course of a LOCA, the primary system pressure drops and the fuel rods heat up. As the rods heat up, the zircaloy cladding experiences a series of phase changes. Thus, the strength of the cladding varies, and at a certain interval during the heatup, the internal pressure within a fuel rod will cause the cladding to plastically deform and swell -- a phenomenon normally referred to as "ballooning."

By this time, the ceramic fuel pellets may be cracked into small pieces. As the cladding swells, the crumbled fuel core drop into and (partially) fill the ballooned region. This same fuel is still producing thermal energy from radioactive decay. Thus, as the fuel settles into a shorter, fatter stack, the local linear power density (kw/ft) increases, even though the total rod power remains constant.

Current, approved ECCS performance analysis codes do not account for fuel settling into ballooned regions. Thus, the lack of inclusion of this effect is a nonconservatism. However, the EG&G study[622] concluded that known conservatisms would more than offset this effect.

Possible Solutions

The only known solution to this possible problem would be to account for the fuel settling and increased kw/ft in the ECCS calculations. This would result in stricter limits on F_Q or MAPLHGR, which would make maneuvering more difficult and which also might result in a plant derate.

PRIORITY DETERMINATION

Frequency Estimate

This item is an issue only for a mitigated LOCA. Therefore, we will add the frequencies of A and S1 LOCAs in WASH-1400.[16] S2 LOCAs will not be included because, if they are mitigated at all, they are so far from the regulatory pellet cladding temperature (PCT) limit of 2200°F that additional heat from fuel crumbling can be accommodated. The sum of the A and S1 frequencies is 4×10^{-4}/RY.

Not all LOCAs are design basis LOCAs. For a design basis LOCA, it is necessary to have the worst break location, the worst break size, the core power distribution at the F_Q (or MAPLHGR) limit, and the worst single failure of the ECCS, plus other conservatisms in initial conditions, system capacities and calculational modeling. The conservatisms in traditional LOCA analyses may well be sufficient to compensate for the effect of fuel crumbling, as was discussed in the EG&G study.[622] However, here we will take the opposite approach and assume no credit for calculation conservatism. Instead we will include the initial conditions in a probabilistic manner.

The probability of a LOCA being a design basis LOCA is very small. However, the worst case is often the worst by only a small margin, i.e. the worst break size and location may be closely followed by other break sizes in other locations. Thus, the probability of a near design basis LOCA is significant. We will assume, based primarily on judgement, that there is at most roughly a 10% probability of a LOCA approaching design basis conditions.

Consequence Estimate

The EG&G study[622] discusses a PBF calculation study. The calculations, which are somewhat conservative, indicated that peak cladding temperatures rose 46°F, and peak centerline temperatures rose 790°F above the nominal (no fuel relocation) case when circumferential strain was 44%. A companion calculation, assuming 89% cladding strain, yielded a cladding ΔT of 406° and a centerline ΔT of 2290°F. However, 89% core-wide cladding strain is not realistic. According to NUREG-0630,[634] a uniform cladding strain of 70% corresponds to all pins swelled into a square shape and pressed against each other with no space remaining to accomodate further swelling. We will use the figures associated with 44% cladding strain, which corresponds to about 78% channel blockage, for core-wide calculations. That is, we will assume that every fuel pin in the core swells 44% throughout its length.

In addition, at some point along each fuel rod, the cladding swells to the point of bursting, which relieves the internal pressure and terminates the ballooning process. Thus, we will further assume that a one foot section of each fuel rod (i.e., about 10% of the length of every fuel rod in the core) swells to 89% cladding strain.

It is the temperature rise which is of concern. Because the degree of ballooning is a function of how the cladding passes through its various phases, rather than only of its peak temperature, the increased temperature is not expected to increase the amount of ballooning. Consequently, fuel crumbling and settling are not expected to result in more flow blockage. Some temperatures of interest are:

3455°F - Eutectic forms (NUREG/CR-1250,[161] Volume 2, Part 2, p. 513)

5148°F - UO_2 melts

During a design basis LOCA, the hottest cladding temperature will be at or slightly below 2200°F before the core is quenched by ECC water. Using the ΔTs calculated for 44% strain, the effect of fuel settling into ballooned regions will raise peak cladding temperatures by 46°F to about 2250°F and peak centerline temperatures 900°F above this to about 3150°F. These figures are well below those needed to cause loss of coolable geometry or release copious quantities of non-volatile fission products or actinides from the fuel matrix via melting or eutectic formation. The only effect is to drive off more noble gases and iodine.

The WASH-1400[16] calculations for Release Categories PWR-9 and PWR-8 (mitigated LOCA with and without containment isolation) assume that 3% of the noble gases and 1.7% of the iodine are released to the containment. We will scale these figures up such that all the noble gases are released. That is, we will assume that the radiological consequences of a fuel pin segment exceeding 2200°F at the cladding are 33 1/3 times those of these two release categories, i.e. 100% of the noble gases and 57% of the iodine are released from fuel which exceeds 2200°F at the cladding. This is, of course, a bounding calculation.

The next question is, given a uniform 46°F increase in cladding temperature throughout the core, what fraction of the core will now exceed the 2200°F licensing limit during the LOCA? Previously, only the hottest point touched 2200°F. Generally, a 10°F change in PCT corresponds roughly to at most a 0.01 change in F_Q for temperature up to 2200°F. This rule of thumb was combined with three-dimensional power distribution information.[633] The calculation is too extensive to be described in detail here. The result was that a uniform 46°F rise will result in roughly 0.015 of the core exceeding the 2200°F limit.

We can now bound the consequences due to core-wide uniform ballooning and fuel setting.

$$\Delta R \text{ (containment isolated)} \leq (0.015)\ (33\ 1/3)\ (120 \text{ man-rem/PWR-9}) \leq 60 \text{ man-rem}$$

$$\Delta R \text{ (containment not isolated)} \leq (0.015)\ (33\ 1/3)\ (75{,}000 \text{ man-rem/PWR-8}) \leq 37{,}500 \text{ man-rem}$$

As discussed earlier, we also assume that a portion of each fuel rod totalling 10% of its length balloons well in excess of 44% strain. If this strain were 89% as in the second PBF calculation, and if the section in question previously just touched the 2200°F PCT limit, the revised peak cladding and centerline temperatures would be roughly 2600°F and 4640°F, respectively. The centerline would not be hot enough to melt and the cladding would not be hot enough to form a eutectic, but, given the roughness of these estimates, the possibility of a copious release of fission products cannot be ruled out. Accordingly, we will not attempt to base a calculation on the degree of ballooning, but instead we will bound the radiological consequences by assuming that these 10% sections of each fuel rod release fission products as if they were molten. (We will not assume any probability of containment overpressure or other mechanistic consequence normally associated with a core-melt, however.) Because 0.1 of the mass of the core is affected, we will use one tenth of the radiological consequences of a PWR-7 release (core-melt with no containment failure) and a PWR-5 release (core-melt with failure of the containment to isolate). The increase in consequences due to enhanced release at the rupture points on each fuel rod is then:

ΔR (containment isolated) $\leq$ (0.1) (2300 man-rem/PWR-7)
$\leq$ 230 man-rem

ΔR (containment not isolated) $\leq$ (0.1) (1,000,000 man-rem/PWR-5)
$\leq$ 100,000 man-rem

Combining all these figures, we can estimate the public risk associated with this issue.

$\Delta FR \leq$ (4 x 10^{-4} LOCAs/RY) (0.10 near design basis LOCAs/LOCA)
x [(0.9 containment isolation/DB LOCA) (60 + 230) man-rem
+ (0.1 isolation failure/DB LOCA) (37,500 + 100,000) man-rem]
$\leq$ 0.56 man-rem/RY

In a 30-year plant lifetime, this is about 20 man-rem.

Cost Estimate

The possible fix for this effect is to reduce F_Q limits by, say 0.05, which would lower the calculated PCT by about 50°F. This will make it harder to maneuver the plant; startups and load changes will take longer. If the plant is or becomes LOCA-limited, this will also cause a derate. In addition, changes in the ECCS analysis generally involve considerable administrative expense. As a minimum cost, we will assume that 5 startups (scram recoveries) a year are extended by one hour, and that one staff-year (including NRC staff time) and $50,000 of computer expense are expended in changing and justifying the technical specifications. The total expense over a 30 year period is thus at least $1M.

Value/Impact Assessment

Based on a total risk reduction of 20 man-rem and a cost of $1M, the value/impact score is given by:

$$S = \frac{20 \text{ man-rem}}{\$1M}$$

$$= 20 \text{ man-rem/\$M}$$

CONCLUSION

The above numbers indicate that this item should be placed no higher than the low priority category. This means that there is insufficient risk-based justification for starting a major re-review of present ECCS Appendix K performance analyses.

However, it should be noted that there are on-going efforts to develop and license ECCS performance models which are more realistic (and consequently less conservative) than the models presently in use. It is not valid to conclude that the effects of fuel crumbling and settling into ballooned regions can necessarily be neglected in any such new, more realistic models. Instead, it is expected that these effects (which are real physical phenomena) will be appropriately addressed in such calculations. Moreover, a separate generic issue on fuel crumbling is not necessary; such work is best done within the scope of the review of the new calculational methodology. It is therefore concluded that this issue should be placed in the LOW priority category.

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ISSUE 93: STEAM BINDING OF AUXILIARY FEEDWATER PUMPS

DESCRIPTION

Historical Background

This issue was recommended[635] for prioritization by DSI after a review of the AEOD engineering evaluation report (AEOD/E325)[636] on vapor binding of the AFW pumps at H. B. Robinson Unit 2. Further AEOD study of the event resulted in recommendations which were documented in AEOD/C404.[637]

The report[637] discusses thirteen occurrences reported in 1983 of steam binding of one or more AFW pumps resulting from the leakage of heated main feedwater into the AFW system. The systems are isolated by various combinations of check valves and control valves. The back-leakage occurred through several valves in series. The heated main feedwater, leaking into the AFW system, flashed to steam in the pumps and AFW discharge lines and resulted in steam binding of the AFW pumps.

Operating experience to date includes 22 events of reported back-leakage in 6 operating PWRs in the USA and at 1 foreign reactor. In other cases, back-leakage has been observed but was not considered as reportable occurrences.

The potential for common mode failure is present whenever one pump is steam-bound because the pumps are connected to common piping with only a single check valve to prevent back-leakage of hot water to the second or third pump. Steam binding of more than one pump was reported to occur in 3 of the 13 events reported in 1983.

Although steam binding of the pumps was reported on only W designed plants, a back-leakage event is believed to have rendered an AFW flow sensor inoperable at Crystal River, a B&W-designed plant. The actual operating status of the pump and train during this event remains unknown. However, the AFW system in all PWRs is sufficiently similar so as to consider it a generic problem for all PWRs.

Safety Significance

The back-leakage of steam represents a potential CCF for the AFW system that could result in the loss of its safety function.

Possible Solutions

AEOD has recommended[637] that regular monitoring of the temperature of the AFW pumps be implemented to provide early detection of back-leakage of main feedwater. This will permit bleeding off the heated water and/or steam before acute steam binding of the pumps can occur. The addition of a pyrometer on the AFW discharge line at or near the pump would permit monitoring of the temperature of the fluid in the system by the plant operators during their routine visual inspections. Records of the temperature readings would show the onset of leakage at an insidious level. Trends of temperature rise times would also provide for the determination of optimal reading and recording intervals which would provide adequate assurance of system availability. The use of a pyrometer would reduce the possibility of error resulting from estimating the temperature by the operator placing his hand close to the auxiliary feedwater pumps or discharge lines.

PRIORITY DETERMINATION

Assumptions

The events experienced in 1983 are considered typical even though the number of events reported annually (prior to 1983) are less. The reporting of back-leakage is only required in those cases in which the pump has been rendered inoperable. Back-leakage which may have been detected on the steam bled from the system before a pump was rendered inoperable might not be considered a reportable occurrence. In fact, it is believed likely that the number of back-leakage events exceeds the number of events reported in 1983 and prior years. However, for this analysis 13 events will be used as the annual occurrence frequency of back-leakage events.

In the calculations, all plants will be assumed to have three auxiliary feedwater pumps although some may have two. The effect of this assumption will be that the total unavailability of the auxiliary feedwater system for those plants having only two pumps will be about 50% lower than the actual unavailability. However, due to the small number of plants having only two pumps, this error is not expected to significantly impact the results.

Frequency/Consequence Estimate

There were 13 events of pump unavailability reported in 1983. Based upon an expected 15 system demands/RY, 3 pumps/plant, and 47 plants, the unavailability/pump-demand (Q) is calculated as follows:

$$Q = 13/(47 \times 3 \times 15) = 6.1 \times 10^{-3}$$

A second pump failure occurring simultaneously was reported to have occurred in 3 of the 13 events. The failure of a second pump is then expected to occur 3 times in 13 events, or at an occurrence rate of 3/13 or 0.23. Assuming that given two pumps having become steam bound, the conditional probability that the third pump will also become steam bound is 0.1 results in a demand unavailability of all 3 AFW pumps of 1.4×10^{-4}.

The original prioritization was based upon the Sequoyah RSSMAP[54] study. This analysis had a TML sequence which led to core melt and the dominant containment failure mode was due to hydrogen burning. It is the belief of many in the PRA risk analysis field that the TML sequence will not lead to core-melt, and that the probability of containment failure due to hydrogen burning may be reduced by orders of magnitude. Further, to assume that the Sequoyah containment (an ice condenser) can be utilized in generic calculations may not be valid. Hence, the consequences were reexamined using the results of the Reactor Safety Study[16] (RSS) and the Surry containment.

In the RSS for Surry, the unavailability of the AFW system was calculated to be 1.5×10^{-4}/demand which did not include steam binding of the AFW pumps. The major sequence affected is the TMLB′ sequence which is increased from 3×10^{-6}/RY to 5.8×10^{-6}/RY by the addition of steam binding to the AFW unavailability. In addition, a very small contribution is made by a TML sequence.

The PWR release categories are as defined in the RSS. The whole body man-rem dose is obtained by using the CRAC code [64] assuming an average population density of 340 persons per square mile (which is the mean for U.S. domestic sites) from an exclusion area of a one-half-mile radius about the reactor out to a 50 mile radius about the reactor. A typical midwest plain meteorology is also assumed. Based upon these assumptions, the public dose resulting from each category is as follows:

Release Category	Dose (man-rem)
1	5.4×10^{6}
2	4.8×10^{6}
3	5.4×10^{6}
5	1.0×10^{6}
6	1.5×10^{5}
7	2.3×10^{3}

The steam binding of the AFW pumps will increase the frequency of the following listed sequences in the categories shown resulting in the listed dose.

Category	Sequence	Frequency Increase (RY^{-1})	Dose (man-rem/RY)
1	TMLB′-α	2.8×10^{-8}	1.5×10^{-1}
2	TMLB′-δ	1.9×10^{-6}	9.12
	TMLB′-γ	6.5×10^{-7}	3.12
3	TML-α	5.6×10^{-8}	3.0×10^{-1}
5	TML-β	2.8×10^{-10}	3.0×10^{-4}
6	TMLB-ε	5.6×10^{-7}	8.4×10^{-2}
7	TML-ε	5.6×10^{-6}	1.3×10^{-2}

Considering only the TMLB′ sequences the resulting dose is 12.5 man-rem/RY. The TML sequences are excluded due to the present uncertainty regarding core-melt of this sequence. For the 90 PWRs which are expected to be operating having an average list of 28.8 years, the total public dose will be 3.2×10^{4} man-rem.

The assumed probability of 0.1 for the third pump failing from steam binding, given that two has so failed, may not be conservative, but rather may be overly optimistic. If it is assumed that of the three events, where 2 pumps were reported to have been steam bound, that one event also involved 3 pumps, then the public dose risk would increase by a failure of 3 to the value of 9.6×10^{4} man-rem.

Cost Estimate

Industry Cost: The cost estimate was based upon a number of engineering assumptions which are believed to be conservatively biased toward the high side of the costs involved. Equipment costs for the pyrometers are estimated to be \$7,500/plant (\$2,500 each); the selection, installation design, ordering, installation and test were estimated to be 10 person-weeks/reactor, or \$22,700. No increase in operating cost is calculated. It is believed that the reading and recording of the temperature of the AFW pumps can be included as part of the plant surveillance activities which are normally accomplished each operating shift. Test and maintenance costs were estimated to be 1 man-week/RY. For the 47 backfit reactors with an average remaining life of 27 years, the maintenance costs total \$2.9M. For the 43 forward-fit reactors having a life of 30 years, the maintenance costs total \$2.9M. It is further estimated that each pyrometer will be replaced twice during the plant life at a cost of \$32,000/plant. The total industry cost to install pyrometers at or near each pump, based upon the above, is \$11.4M.

NRC Cost: The NRC cost is estimated to not exceed 1 man-week/reactor or \$0.2M for all affected plants.

Value/Impact Assessment

(1) For the scenario in which the probability of the third pump failing steam bound is 0.1, the value/impact score is given by:

$$S = \frac{3.2 \times 10^4 \text{ man-rem}}{\$(11.4 + 0.2)\text{M}}$$

$$= 2.8 \times 10^3 \text{ man-rem/\$M}$$

(2) For the scenario in which the probability of the third pump failing steam bound is 0.33, the value/impact score is given by:

$$S = \frac{9.6 \times 10^4 \text{ man-rem}}{\$(11.4 + 0.2)\text{M}}$$

$$= 8.4 \times 10^3 \text{ man-rem/\$M.}$$

CONCLUSION

Both the total dose in man-rem and the value/impact score vary from bordering between medium-to-high priority for the third pump failure of 0.1, to high of the third pump failure if the third pump failure were 0.33. In light of the uncertainty associated with this issue a HIGH priority is assigned.

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SECTION 4

HUMAN FACTORS ISSUES

The investigations that followed the TMI-2 accident identified the need to incorporate human factors considerations into the regulations and guidance governing the design and operation of nuclear power plants. NUREG-0660[48] described a number of tasks that were to be performed by the nuclear industry and NRC. A significant number of these tasks focused upon improving nuclear power plant safety through increased attention to the human element. Considerable progress has been made on many of the NUREG-0660[48] tasks.

The issues presented in this section include those outlined in the Human Factors Program Plan (HFPP) and documented in NUREG-0985, Revision 1.[651] This plan describes the human factors-related work required to complete the NUREG-0660[48] human factors tasks as well as the additional human factors-related efforts, identified during implementation of NUREG-0660[48] tasks, that require NRC attention.

The lead responsibility and a summary of the findings for each item listed in this section can be found in Table II of the Introduction.

ITEM HF01: HUMAN FACTORS PROGRAM PLAN (HFPP)

DESCRIPTION

The many investigations which followed the accident at TMI-2 identified the need to bring human factors considerations into the requirements and regulations for the design, construction, and operation of nuclear power plants. NUREG-0660[48] contains many items which address human factors concerns. The "U.S. Nuclear Regulatory Commission Human Factors Program Plan," NUREG-0985,[603] was subsequently developed to provide a more current and comprehensive consideration of the outstanding human factors issues related to the design, operation, and maintenance of nuclear power plants. This plan describes: (1) the technical assistance and research activities planned to provide the technical bases for the reduction of the human-factors-related tasks described in NUREG-0660,[48] and (2) the additional human factors efforts identified during implementation of the TMI Action Plan that should receive NRC attention. The HFPP identifies seven program elements:

(1) Staffing and Qualification
(2) Training
(3) Licensing Examinations
(4) Procedures
(5) Management and Organization
(6) Man-Machine Interface
(7) Human Reliability

A more detailed description of each element of the program can be found in Appendix HF01 to this prioritization.

PRIORITY DETERMINATION

The prioritization analysis was performed by PNL under contract to NRC. Details of the PNL analysis will be included in NUREG/CR-2800.[64]

To ascertain the risk reduction achievable by the elements in the plan, PNL polled a group of specialists representing various disciplines in the human factors and plant operations skills. These experts estimated the achievable reduction in human error rate that could be realized by the implementation of regulatory requirements and guidance which would come about as the solution of each individual program element.

Element (7), Human Reliability, is viewed as an issue that does not directly relate to protecting the public health and safety or the environment. Rather, it is related to increasing the knowledge, certainty, and understanding of the other human factors issues in order to increase confidence in assessing levels of safety and improving or maintaining the NRC capability to make independent assessments of safety. As a result, this element is a licensing issue.

Since the plan was developed specifically to address the interrelationship of human factors issues, the first six elements of the HFPP cannot be treated as mutually-exclusive elements. For example, the adequacy of procedures is affected by training, that is, training assists in the proper interpretation and understanding of procedures. To account for this interdependency or overlap of the individual elements, PNL used four separate approaches to calculate the contribution of each element to a reduction in risk. Each of these approaches accounts for overlaps by compiling the results of the questionnaire taken in the poll in a different way. No one approach is believed to accurately model the real overlap that exists between the various plan elements. The first approach simply considered that there were no overlaps between the various plan elements. The second approach considered that all overlaps of other elements (on the first element) were removed from the first element, and then all overlaps (of the first element) on the other elements were added to the remaining direct effect of the first element. The third approach considered overlap removal and addition based on an initial order of implementation. The fourth approach considered overlap removal and addition based on the most incrementally efficient order of implementation. The reevaluation occurred after each element was implemented. These four approaches were intended to consider the various ways that element overlaps can be accounted for and how these ways affect improvement of human performance assessments.

Frequency/Consequence Estimate

The results of the questionnaire were used to calculate the risk change using the Oconee and Grand Gulf RSSMAP studies.[54] The overall reduction in human error value was estimated to be 60 percent. This is broken down into the six categories and models as shown in the interim results, Table HF01-1. This overall value was applied to all operator error parameters in the Oconee and Grand Gulf risk assessments and the resulting risk changes were calculated. A risk change was calculated for each of the four dependency models identified by A, B, C, and D. This risk change value is the total risk reduction which would result if all of the six plan elements were implemented. This risk change was multiplied by the individual elements rating to yield the risk change which could be attributed to each element by each dependency model as shown in Table HF01-2. Based on 90 PWRs and 44 BWRs with average lives of 28.8 years and 27.4 years, respectively, the total industry risk reduction by approach is given in Table HF01-3.

Cost Estimate

The estimated costs involved in the HFPP were derived using the previously estimated costs of the human factors generic safety issues in the TMI Action Plan.[48] Each of the individual generic safety issues were examined to determine which issues were applicable to the program plan and assigned a percentage application to the HFPP and to specific HFPP elements as shown in Table HF01-4.

Safety issue costs for NRC and utility development, implementation, and operation were examined and assigned to the different elements in the percentage weights previously determined. Costs associated with improving the plant maintenance program were removed since the maintenance improvement activity is to be prioritized as a separate program. However, maintenance costs (ongoing element costs) associated with keeping the element functional are included.

TABLE HF01-1

INTERIM RESULTS BASED ON COMPILED QUESTIONNAIRE ANSWERS

ELEMENT AREAS	RATINGS (%) BY APPROACH A	B	C	D
Staffing and Qualifications	11.1	15.9	7.9	0.9
Training	22.3	21.7	32.8	41.6
Licensing Examinations	11.4	10.4	3.0	6.7
Procedures	18.6	16.1	17.1	9.0
Man-Machine Interfaces	17.7	10.9	3.4	14.9
Management and Organization	18.9	25.1	35.8	26.9
TOTAL:	100.0	100.0	100.0	100.0

TABLE HF01-2

RISK CHANGE RESULTS BASED ON INTERIM RATING RESULTS

	RISK REDUCTIONS (MAN-REM/RY) BY APPROACH AND REACTOR TYPE							
	A		B		C		D	
ELEMENT AREAS	PWR	BWR	PWR	BWR	PWR	BWR	PWR	BWR
Staffing and Qualifications	6.2	4.8	8.9	6.8	4.4	3.4	0.5	0.4
Training	12.5	9.6	12.2	9.3	18.4	14.1	23.3	17.9
Licensing Examinations	6.4	4.9	5.8	4.5	1.7	1.3	3.8	2.9
Procedures	10.4	8.0	9.0	6.9	9.6	7.4	5.0	3.9
Man-Machine Interfaces	9.9	7.6	6.1	4.7	1.9	1.5	8.3	6.4
Management and Organization	10.6	8.1	14.1	10.8	20.0	15.4	15.1	11.6
TOTAL:	56.0	43.0	56.1	43.0	56.0	43.1	56.0	43.1

TABLE HF01-3

TOTAL INDUSTRY RISK CHANGE RESULTS

ELEMENT AREAS	RISK REDUCTIONS BY APPROACH (10^4 MAN-REM) A	B	C	D
Staffing and Qualifications	2.2	3.1	1.6	0.18
Training	4.4	4.3	6.5	8.2
Licensing Examinations	2.2	2.0	0.6	1.3
Procedures	3.7	3.2	3.4	1.8
Man-Machine Interfaces	3.5	2.1	0.67	2.9
Management and Organization	3.7	5.0	7.0	5.3
TOTAL:	20	20	20	20

TABLE HF01-4

ALLOCATION OF TMI ISSUES IMPACT AMONG HUMAN FACTORS PROGRAM ELEMENTS

Issue Number	Title	%H*	% Scope SQ	T	LE	P	MO	MMI
STAFFING AND QUALIFICATIONS (SQ)								
I.A.1.4	Operating Personnel and Staffing Long-Term Upgrades (Resolved)	-	0	0	0	0	0	0
I.A.2.2	Training and Qualifications of Operating Personnel	50	50	25	25	0	0	0
I.A.3.3	Requirements for Operator Fitness	100	90	0	10	0	0	0
TRAINING (T)								
I.A.2.6(4)	Long-Term Upgrade of Training and Qualifications (Training Workshops)	100	0	100	0	0	0	0
I.A.2.6(6)	Nuclear Power Fundamentals for Operator Training (Resolved)	-	0	0	.10	0	0	0
I.A.2.6.(1,2,3,5)	Simulators	100	57	0	10	10	0	5
I.A.2.7	Accreditation of Training Institutions	100	10	80	10	0	0	0
I.A.2.4	NRR Participation in Inspector Training	30	0	100	0	0	0	0
I.A.4.2	Long-Term Training Simulator Upgrade	100	0	50	0	25	0	25
LICENSING EXAMINATIONS (LE)								
I.A.3.4	Licensing of Additional Operations Personnel	25	20	0	70	0	10	0
I.A.3.2	Operator Licensing Program Changes (Resolved)	-	0	0	0	0	0	0
PROCEDURES (P)								
I.C.9	Long-Term Program Plan for Upgrading Procedures	100	0	15	15	70	0	0
I.C.1(4)	Confirmatory Analysis of Selected Transients	100	0	15	15	70	0	0
MANAGEMENT AND ORGANIZATION (MO)								
I.B.1.1(1,2,3,4,6)	Management for Operations: Organization and Management of Long-Term Improvements	40	10	10	5	15	60	0
II.J.3.1/II.J.3.2	Organization and Staffing to Oversee Design Construction	0	25	25	0	0	50	0
MAN-MACHINE INTERFACE (MMI)								
I.D.3	Safety System Status Monitoring	100	0	10	0	10	0	80
I.D.4	Control Room Design Standard	100	0	10	0	10	0	80
I.D.5(3,4,5)	Control Room Design: Improved Control Instrumentation Research	100	0	10	0	10	0	80

* Percentage of TMI Action Plan items in the HFPP:

- Issue I.A.2.2 scope includes maintenance and technician training, so it is assumed that only 50% of the costs identified are applicable to the other portions of the HFPP.

- Issue I.A.2.4 is primarily a licensing issue; only 30% actually addresses human factors operating issues.

- Issue I.A.3.4 impacts primarily the licensing of personnel associated with maintenance operations. It is assumed that only 25% of the scope/costs apply to the HFPP.

- Issue I.B.1.1(1,2,3,4,6) addresses issues classified as maintenance; hence, only 40% is assigned to the HFPP.

- Issues II.J.3.1/II.J.3.2 are addressed towards improving management oversight during design and construction and, hence, have no connection with the HFPP.

The combined scope of all the previous human factors safety issues in the TMI Action Plan[48] was examined to see if it gave a comprehensive coverage of the stated goals of the individual HFPP elements. The possibility of duplication of effort among issues must also be considered.

As a first estimate, it was judged that the previous safety issues do in fact give a sufficient coverage of scope such that costs will be representative for the Staff and Qualification, Training, Licensing Examination, and Man-Machine Interface (MMI) program elements. Excessive overlap between issues was not readily apparent.

The Procedures element was covered by only two previous issues. However, its role in further development with training, and especially advanced control room design (MMI) was recognized. As a first estimate, the values in Table HF01-4 will be used.

The Management and Organization element was judged to be lacking in coverage of scope as represented by the safety issues. Item II.J.3.1 especially had no input to this program; however, an issue could be drafted along similar lines to ensure consideration of human factors associated problems in management circles during stages of design, construction, and modification. An approach similar to Item I.A.3.3 which involves NRR human factors input to OIE inspector training could also be applied to the training of engineers undergoing training to oversee design and construction. These engineers would then play a strong management role after construction, as with Issue II.J.3.1. This area has also been recently recognized as one where substantial commitment and improvement on the part of utilities may be needed. As a result of this, it will be assumed that the costs identified by Issue II.J.3.1 will also be included here as representative of a similar type of program for increasing management awareness of human factors requirements in their organizational structure.

Table HF01-5 presents estimates of the industry costs to implement the HFPP actions. Table HF01-6 presents estimates of the NRC costs to establish regulatory requirements and to provide for annual review. The annual costs for the operation and maintenance of each program element was multiplied by 28.3 years, the average remaining plant life, to determine the lifetime costs.

Implementation of the HFPP elements is expected to result in cost savings due to a reduction in downtime and increased plant availability. However, the impact on plant availability could be expected to be more difficult to quantify the further the program moves away from hardware. Hence, while it is recognized that a cost saving will result from the implementation of the findings in the HFPP, the costs were not calculated.

Value/Impact Assessment

The value/impact ratios for each element and the total plan are presented in Table HF01-7. The value/impact assessment is shown for each of the four approaches used to calculate the risk reduction. As shown in this table, the element scores range from 1.2 to 96 man-rem/$M. The value/impact assessment for the entire plan was calculated to be 27 man-rem/$M.

TABLE HF01-5

ESTIMATES OF INDUSTRY COSTS FOR HUMAN FACTORS PROGRAM

Program Element	Implementation ($M)	Op. & Maintenance ($M/yr)	Op. & Maintenance Total ($M)	Total Element Cost ($M)
Staffing & Qualifications	82.9	50.4	1400	1480
Training	452.4	81.0	2300	2750
Licensing Examinations	47.5	20.1	570	620
Procedures	219.2	28.5	800	1020
Man-Machine Interface	384.3	10.6	300	680
Management & Organization	33.1	23.1	650	680
TOTAL:	1200.0	210	6000	7200

TABLE HF01-6

ESTIMATES OF NRC COSTS FOR HUMAN FACTORS PROGRAM

Program Element	Development ($M)	Implementation ($M)	Op. & Maintenance ($M/yr)	Op. & Maintenance Total ($M)	Total Element Cost ($M)
Staffing & Qualifications	1.8	0.6	0.9	25.0	27.4
Training	1.3	1.9	1.1	31.1	34.6
Licensing Examinations	5.8	1.1	1.3	36.8	43.7
Procedures	2.2	1.2	0.7	19.8	23.2
Man-Machine Interface	1.9	1.6	0.15	4.2	7.7
Management & Organization	1.4	0.4	1.8	52.0	53.6
TOTAL:	14.4	6.8	5.9	169.0	190.0

TABLE HF01-7

SUMMARY OF RESULTS

		HFPP ELEMENT						
		Staffing Qualifications	Training	Licensing Examination	Procedures	Man-Machine Interface	Management Organization	Total
Public Risk (10^4 man-rem)	A -	2.2	4.4	2.2	3.7	3.5	3.0	20
	B -	3.1	4.3	2.0	3.2	2.1	5.0	20
	C -	1.6	6.5	0.6	3.4	0.67	7.0	20
	D -	0.18	8.2	1.3	1.8	2.9	5.3	20
Total Cost Industry & NRC ($M)		1500	2800	660	1000	690	730	7400
Value/Impact Assessment (man-rem/$M)	A -	15	16	33	37	51	51	27
	B -	21	15	30	32	30	69	27
	C -	11	23	9.1	34	9.7	96	27
	D -	1.2	29	20	18	42	73	27

Other Considerations

The risk reduction calculation resulting from human error reduction considers only the reduction in human errors after the assumed initiation of an accident sequence. The PRA analyses on which these calculations are based do not give credit to a reduction in transient events due to improved operator capability, nor do they give credit for operator intervention to overcome hardware malfunctions. Further, the existing PRAs on which this analysis was based, in general, considered only the type of human errors classified as procedural errors. These acts are also often called the "skill and rule" based actions. Neglected in these PRA analyses are the cognitive acts of the operator which may be the most crucial. Thus, the contribution to risk reduction resulting from reducing human error is underestimated. The low value/impact assessments emphasize the necessity of carefully selecting regulatory requirements which will be most cost effective.

CONCLUSION

The HFPP as well as most of its major elements should be rated high priority. A few of the elements depending upon the relative contribution of the other associated elements might result in a medium priority assignment. However, the uncertainty associated with the dependency calculations is sufficiently uncertain as to warrant assigning the major elements a high priority rating. Hence, a HIGH priority ranking is assigned the HFPP and its major plan elements except HF01-7 which is a licensing issue.

REFERENCES

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980.

54. NUREG/CR-1659, "Reactor Safety Study Methodology Application Program," U.S. Nuclear Regulatory Commission, 1981.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

603. NUREG-0985, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," U.S. Nuclear Regulatory Commission, August 1983.

651. NUREG-0985, Revision 1, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," U.S. Nuclear Regulatory Commission, September 1984.

APPENDIX HF01

HUMAN FACTORS PROGRAM PLAN

TASK HF01.1.0: STAFFING AND QUALIFICATIONS

This task was developed to assure that the number and capabilities of the staff at nuclear power plants are adequate to provide a safe operation. To meet this goal, consideration will be given to: (1) the numbers and functions of the staff needed to safely perform all required plant operations, maintenance, and technical support for each operational mode; (2) the minimum qualifications of plant personnel, in terms of education, skill, knowledge, training, experience, and fitness for duty; and (3) appropriate limits and conditions for shift work, including overtime, shift duration, and shift rotation.

ITEM HF01.1.1: NPP STAFFING REQUIREMENTS

This item will address the following:

(a) The NRC will determine the minimum appropriate shift crew staffing composition. This determination will be made from developed personnel projection and allocation models and from evaluations of job and task analysis and probabilistic risk assessment data. Current staffing practice of both foreign and domestic utilities were surveyed to evaluate current practices, regulations and current staffing levels, considering such variables as plant size, control room arrangement and configuration, and plant layout. A rule for inclusion in 10 CFR Part 50, §§50.54(m)(2) was prepared regarding licensed operator staffing. A review of SRP[11] Section 13.1.3 which includes staffing will be developed.

(b) The need for engineering expertise on shift will be decided. This decision will be based in part upon the functions and duties required using the results of job/task analysis and evaluation of the current shift technical advisor experience. Consideration will also be given on how best to incorporate this expertise into the plant crew compliment. A proposed rule for 10 CFR 50 has been prepared and a final policy statement on the inclusion of engineering expertise on shift has been developed.

ITEM HF01.1.2: NPP PERSONNEL QUALIFICATIONS REQUIREMENTS

This item will address the following:

(a) The minimum training, education, and experience requirements for shift operating crews will be determined from a review of job and task analysis data. The relationship between education, training, and experience will be assessed and the trade-offs among these related factors determined. A rule for 10 CFR 50 will be prepared on minimum crew qualifications.

(b) A study will be done of the feasibility and value of licensing or certifying nuclear power plant personnel other than operators. A rule on degree requirements for the operating staff will be prepared.

ITEM HF01.1.3: GUIDANCE ON LIMITS AND CONDITIONS OF SHIFT WORK

Experience and research data indicate that shift work and the use of overtime can have an adverse effect upon operator performance. To determine the appropriate limits and conditions for shift work, activities are planned: (1) to determine the effects of varying shift duration using nuclear power plant simulators, and (2) to survey and assess the experience of other industries with job requirements similar to the nuclear industry with regard to shift arrangements and rotation. This effort will allow the NRC to establish trade-offs among factors affecting shift work and overall safe performance requirements. The results are to be reported as a NUREG document and a specific research effort will be undertaken if shift rotation and conditions of overtime are found to be serious human factors problems.

ITEM HF01.1.4: FITNESS FOR DUTY

A proposed rule, revising 10 CFR Part 73, relating to fitness for duty for personnel having access into nuclear power plants or involved with their operation has been prepared.

TASK HF01.2.0: TRAINING

The task was developed to provide assurance that personnel are able to meet job performance requirements, that training properly accounts for pertinent safety issues, and that a mechanism exists for upgrading and assuring the quality of training programs. The Nuclear Waste Policy Act of 1982, Section 306 of PL 97-425, directs the NRC to promulgate regulations and regulatory guidance for the training of nuclear power plant personnel. Areas to be addressed include simulator training requirements, operator requalification programs, team training, instructional requirements for training programs, and the administration of examinations. The planned activities in this task and its end products are supportive of this law.

ITEM HF01.2.1: DEVELOPMENT OF TRAINING REGULATION AND GUIDANCE

NRR will perform the activities required to develop regulations and guidance for the training and qualifications of civilian nuclear power plant operators, supervisors, technicians, and other appropriate operating personnel. Proposed regulations in 10 CFR Part 50, §§50.200 - 50.250 direct the utilities to review their training programs to assure the use of a systematic approach to training. The purpose of this effort is to focus the utility training program on the knowledge, skills and abilities required to operate the nuclear power plant safely and assure the licensed operators' ability to respond correctly to unexpected events. As appropriate, this program will recognize the relationship to INPO's accreditation activity and INPO's effort to develop a handbook on application of a systematic approach to training. This activity will result in a regulation for the training and qualification of nuclear power plant personnel. Regulatory Guides will accompany the proposed training regulation. A NUREG will be prepared to provide guidelines for team training.

ITEM HF01.2.2: NRC TRAINING EVALUATION PROGRAM

Criteria will be developed to allow consistent and objective inspections to determine compliance with the training regulation. A regulatory position is to be established regarding accreditation of training programs. This activity will also result in development of criteria to evaluate the qualifications for training instructors. Revisions to Chapter 13.2 of NUREG-0800, "Standard Review Plan," will be prepared. Inspection modules for assessing training programs will be prepared or modified for use by the Office of Inspection and Enforcement and the Regional Offices, based on the training review criteria developed earlier.

TASK HF01.3.0: LICENSING EXAMINATION

One purpose of this task is to ensure that the licensing examination for reactor operators and senior operators is a valid measure of the operator's knowledge and ability to perform the necessary tasks and functions required to safely operate and control commercial nuclear power plants. The second purpose is to ensure that examinations are administered in a consistent manner by the various NRC examiners to enhance reliability and efficiency. The intent is to perform these modifications to the examination process without unnecessary impact to current license candidates and training programs.

ITEM HF01.3.1: THE EXAMINATION CONTENT

A catalog of the reactor operator and senior operator tasks and duties, and the required knowledge, skills and abilities necessary for safe performance will be formulated using available generic job and task analyses. A computerized bank of examination questions for use in test construction and examination validation will be developed and updated using this catalog.

Additionally, test specifications will be developed for licensing examinations to provide examination plans which outline the necessary types of knowledge required to be assessed during examinations. An evaluation of the feasibility of identifying or developing on-the-job performance measures which can be used in assessing the ability of the examination process to predict operator performance will be conducted. Long-term examination development/validation strategies will be developed based upon the results of current examination modifications and content validation.

ITEM HF01.3.2: THE EXAMINATION PROCESS

To increase the efficiency, reliability, and validity of the licensing examination process, the DHFS will evaluate new examination procedures. These new procedures will take into consideration the problems and issues associated with the current examination process from the examiner, candidate and utility perspectives. The examination process and practices of similar applicable agencies and organizations will be reviewed. The input from industry training staff and reactor operators regarding problems or issues underlying the current licensing examinations will be solicited. The results will be the identification of improvements to optimize the format and procedures relating to written, oral and simulator examinations. From this identification activity, standardized examination practices and guidelines will be developed. The test examiners will also be trained on test development, administration and grading techniques to assure consistency and reliability. A revision to 10 CFR Part 55 will be prepared to reflect changes made in the examinations and in the examination process. Revision to Regulatory Guide 1.149, "Nuclear Power Plant Simulators for Use in Operator Training," will also be accomplished.

TASK HF01.4.0: PROCEDURES

This task is to provide assurance that plant procedures are adequate and can be used effectively. The objective is to provide procedures which will guide the operators in maintaining the plant in a safe state under all operating conditions, including the ability to control upset conditions without first having to diagnose the specific initiating event. This objective is to be met by: (1) developing guidelines for preparing, and criteria for evaluating emergency operating, normal operating, and the other procedures which affect plant safety; and (2) upgrading the procedures, training the operators in their use, and implementing the upgraded procedures.

ITEM HF01.4.1: PROCEDURES GUIDANCE AND CRITERIA

This item will address the following:

(a) Guidance will be developed which will provide the necessary instructions to prepare improved emergency operating procedures, abnormal operating procedures, normal operating procedures, maintenance procedures, and procedures for emergency plan implementation, refueling, administration, safeguards, and security. Generic technical guidance is to be provided by industry; the NRC and industry will jointly coordinate the development of human factors guidelines. Research will be accomplished to develop methods and evaluate alternate techniques and formats for the display of procedures, e.g., computerized CRT presentation. Results of these activities will be published as NUREG reports.

(b) Criteria to evaluate and audit emergency operating procedures by the regions will be prepared by NRR and IE. This criteria will be published as a revised inspection module. Similar criteria and inspection modules will be developed when the guidelines for the upgrading of other procedures are completed.

TASK HF01.5.0: MAN-MACHINE INTERFACE (MMI)

The objective of this task is to ensure that the MMI is adequate for the safe operation and maintenance of nuclear power plants. This objective will be attained by developing: (1) human factors engineering guidelines for correcting man-machine interface problems, and (2) regulatory guidance for integrating human factors engineering into new designs and into advanced technological improvements incorporated into existing designs. This activity will also provide for the preparation of evaluation tools for: (1) the next generation of nuclear power plant; and (2) for expected changes or upgrading to designed plants in the area of data and information management and improved annunciator systems. In addition, these efforts will improve the staff's capability to evaluate reactor incidents involving man-machine interface errors.

ITEM HF01.5.1: MMI GUIDANCE FOR EXISTING DESIGNS

The regulatory efforts to date dealing with the MMI have been limited to the control room and the remote shutdown panel. Further guidance is necessary regarding local control stations and auxiliary operator interfaces. Additional guidance may also be required regarding improvements to the existing annunciator system.

(a) Information will be developed to determine if guidance on local control station design and auxiliary operator interfaces with these stations is required. To accomplish this subtask, job/task analyses of control room crew activities will be conducted to identify and describe communication and control links bewteen the control room and the auxiliary control stations. In addition, the functions of the auxiliary personnel will be analyzed from the task analyses to estimate the potential impact of auxiliary personnel job errors on plant safety. NUREG reports will be published to report the findings.

(b) The information provided in NUREG-0700, "Guidelines for Control Room Design Reviews," provides the guidance, which if incorporated, should minimize the potential for human errors associated with these systems. However, some of these standards are difficult to apply except as long-term design changes. Guidance will be developed for near-term improvements which address the techniques for implementing the quality standards of NUREG-0700. An assessment of the impact of NUREG-0700 guidelines on operating control rooms will be performed to identify if revisions are needed to NUREG-0700.

ITEM HF01.5.2: MMI GUIDANCE FOR DESIGNS BASED ON ADVANCED TECHNOLOGIES

The existing human engineering guidelines for nuclear power plant control rooms primarily address the control, display and information concepts and technologies which are now being used in process control systems. While these guidelines are adequate for the current generation of nuclear power plants, they may not be sufficient for advanced and developing technologies which may be introduced into existing and future designs. This concern is addressed by the following activities:

(a) Computers - A program plan will be developed to evaluate the safety signficance and problems relating to the management of data and information in the nuclear power plant control room during abnormal events. Products may include the development of guidelines on control room information management during severe transients and accidents. These guidelines may be in the form of NUREG reports and Regulatory Guides.

(b) Advanced Controls and Displays - Staff guidance pertinent to the man-machine interface involving new control and display techniques will be prepared. These guidance documents will require: (1) the identification of new developing display and control technologies having a potential application in nuclear power plant control rooms, (2) development of evaluating methods and design criteria related to visual displays, and (3) establishing the criteria needed for regulatory assessment of advanced control room concepts. In addition, the control and display requirements for crew response needs following a seismic event will be identified.

(c) Function Allocation - An integrated program plan for the investigation of function allocation will be prepared. The plan will address: (1) the identification of nuclear power plant functions involving the human, (2) whether the current function allocations permit the reliable performance of functions assigned to humans, (3) the need to reallocate functions between the human and the machine, (4) which functions should be reallocated, and (5) the identification of those design changes which enhance function performance. In addition, the plan will address the feasibility and desirability of applying cognitive workload measurement techniques to a selected list of operator functions.

(d) Advanced Annunciator Systems - Improved annunciator systems are expected to become available which will utilize advanced technologies. Guidelines for the utilization and evaluation of these longer-term annunciator improvements will be developed. These guidelines will be based upon evaluations of results from advanced concept activities being performed by governmental and commercially sponsored research activities.

(e) Safety Status Indication - Based upon the results of a current project investigating means for monitoring and verifying operations, tests, and maintenance activities, the staff will make determinations concerning: (1) the comparative adequacy of status monitoring in plants that do not have automatic monitoring systems, (2) the adequacy of operational systems designed to be in conformance with Regulatory Guide 1.47, and (3) the development of long-term improvement guidance addressing the feasibility and value/impact of instrumentation backfits.

TASK HF01.6.0: MANAGEMENT AND ORGANIZATION

The objective of this effort is to ensure that utility management and organization is adequate to provide for safe operation of their nuclear power plants. This objective will be accomplished by: (1) developing approaches and techniques to optimize relationships between utility organization and management factors and plant safety, (2) developing and testing reliable, objective evaluation procedures for assessing the adequacy of organization and management functions, and (3) providing a sound technical basis for an NRC regulatory position on organization and management at operating nuclear power plants.

ITEM HF01.6.1: REGULATORY POSITION ON MANAGEMENT AND ORGANIZATION AT OPERATING REACTORS

This issue will be resolved by accomplishing the following objectives: (1) complete evaluation of existing management and organization assessment activities such as INPO assistance visits, NRC Performance Assessment Team (PAT) Inspection, and Systematic Assessment of Licensee Performance Program (SALP); (2) develop technical basis for an NRC regulatory position on management and organization at operating nuclear power plants; (3) make decision on need for a new regulatory position on management and organization; and (4) develop new regulatory position.

ITEM HF01.6.2: NRC MANAGEMENT AND ORGANIZATION GUIDELINES AND ASSESSMENT PROCEDURES FOR OPERATING LICENSE REVIEWS

This issue will be resolved by accomplishing the following objectives: (1) develop guidelines which describe required minimums for acceptable management and organization at nuclear power plants, (2) prepare workbook to accompany guidelines, (3) publish the guidelines and workbook for use by NRR reviewers of operating license applications, and (4) train technical reviewers to effectively use workbook and improve interview skills.

TASK HF01.7.0: HUMAN RELIABILITY

The primary purposes of this element are to develop a technical support system for NRC reliability evaluations, especially the PRA programs, and to provide feedback links from operating experience to other elements of the human factors program. A secondary goal is to develop approaches for employing human error data as baseline performance measures in man-machine safety system evaluations.

ITEM HF01.7.1: HUMAN ERROR DATA ACQUISITION

Activities ongoing and planned are designed to provide NRC reliability evaluation programs with methods and techniques for acquiring reliable human error data from a variety of nuclear power related sources. Significant research involves developing guidelines for acquiring human error data from expert judgment, training simulators, operating nuclear power plants using LER data, and from a non-punitive reporting concept.

ITEM HF01.7.2: HUMAN ERROR DATA STORAGE AND RETRIEVAL

Activities are designed to provide the NRC with a human reliability data bank for use in processing human error data for use by reliability evaluation specialists. Planned activities include developing methods and procedures for computing human error probability statements from diverse information sources and storing, updating and retrieving human error probability statements and related information.

ITEM HF01.7.3: RELIABILITY EVALUATION SPECIALIST AIDS

A comprehensive and accurate analysis of human behavior sequences leading to recognition, diagnosis and reaction to nuclear power station normal, transient and emergency events is necessary for risk assessment. Analytic techniques and methods for portraying adequately the human segments of those events are needed, especially events involving redundant or interdependent actions by individuals or groups. Also needed are techniques for analyzing cognitive and performance shaping factor (e.g., stress, fatigue, attitude) aspects of human behavior. Significant research activities in this area involve: 1) developing techniques for analyzing safety-related events, especially those involving redundancy and/or interdependent actions; and 2) investigating the feasibility of objectively analyzing cognitive and performance shaping aspects of human behavior within the content of NRC reliability evaluation programs, especially PRAs.

ITEM HF01.7.4: SAFETY EVENT ANALYSIS RESULTS APPLICATION

The PRAs are a potential source of quantitative and qualitative human performance data, both generic and plant-specific. Human reliability research will be directed toward developing and testing approaches and techniques for systematically using human performance data from PRAs to: (1) identify generic and plant-specific man-man and man-machine safety system retrofit requirements, (2) establish objective baseline performance measures for evaluating plant retrofits, and (3) identify future human reliability/human factors research needs.

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The report presents the priority rankings for generic safety issues related to nuclear power plants. The purpose of these rankings is to assist in the timely and efficient allocation of NRC resources for the resolution of those safety issues that have a significant potential for reducing risk. The safety priority rankings are HIGH, MEDIUM, LOW, and DROP and have been assigned on the basis of risk significance estimates, the ratio of risk to costs and other impacts estimated to result if resolutions of the safety issues were implemented, and the consideration of uncertainties and other quantitative or qualitative factors. To the extent practical, estimates are quantitative.

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